QA: N/A

Yucca Mountain Site Characterization Project

Technical Guidance Document for License Application Preparation YMP/97-03

Revision 1

September 1999

U.S. Department of Energy
Office of Civilian Radioactive Waste Management
Las Vegas, Nevada

QA: N/A

Yucca Mountain Site Characterization Project

Technical Guidance Document for License Application Preparation YMP/97-03

Revision 1

September 1999

U.S. Department of Energy
Office of Civilian Radioactive Waste Management
Las Vegas, Nevada

Preparation:

S.J. Brocoum, Acting Assistant Manager
Office of Licensing and Regulatory Compliance
Yucca Mountain Site Characterization Office

Date

Approval:

J.R. Dyer, Project Manager
Yucca Mountain Site Characterization Office

Date

R. W. Clark, Acting Director
Office of Quality Assurance
Yucca Mountain Site Characterization Office

Date

CHANGE HISTORY

REV. NO.	ICN NO.	EFFECTIVE DATE	DESCRIPTION OF CHANGE
0		08/24/98	Initial Issue
1			Partial revision to replace 10 CFR 60, Energy: Disposal of High-Level Radioactive Wastes in Geologic Repositories, requirements with proposed 10 CFR 63, Energy: Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, Nevada, requirements; to update guidance regarding codes, standards, and U.S. Nuclear Regulatory Commission Key Technical Issues, and DOE Interim Guidance dated June 18, 1999, pending issuance of new U.S. Nuclear Regulatory Commission (NRC) Regulations for Yucca Mountain, Nevada; and to revise level of detail to reflect the Project White Paper titled "Criteria for Design Information Needed for the License Application for Construction Authorization."

INTENTIONALLY LEFT BLANK

CONTENTS

Page

CONTENTS (Continued)

Page

CONTENTS (Continued)

Page

CONTENTS (Continued)

Page

CONTENTS (Continued)

Page

CONTENTS (Continued)

Page

CONTENTS (Continued)

Page

CONTENTS (Continued)

Page

CONTENTS (Continued)

Page

INTENTIONALLY LEFT BLANK

ACRONYMS

ALARA	as low as is reasonably achievable
CA	construction authorization
CFR	Code of Federal Regulations
CRWMS	Civilian Radioactive Waste Management System
DBE	design basis event
DOE	U.S. Department of Energy
EBS	engineered barrier system
EIS	environmental impact statement
ESF	Exploratory Studies Facility
FEP	feature, event, and process
GROA	geologic repository operations area
HEPA	high-efficiency particulate air
HLW	high-level radioactive waste
IRSR	issue resolution status report
ISA	integrated safety analysis
KTI	key technical issue
LA	License Application
LLW	low-level radioactive waste
M&O	Management and Operating Contractor
MGR	monitored geologic repository
NRC	U.S. Nuclear Regulatory Commission
NWPA	Nuclear Waste Policy Act of 1982, as amended
OCRWM	Office of Civilian Radioactive Waste Management
PA	performance assessment
QA	quality assurance
QARD	*Quality Assurance Requirements and Description*
QL	quality level
R&D	research and development
SAR	Safety Analysis Report
SDD	system description document
SI	International System of Units
SNF	spent nuclear fuel
SRR	Site Recommendation Report
SSC	structure, system, and component
SZ	saturated zone

ACRONYMS (Continued)

TEDE	total effective dose equivalent
TGD	technical guidance document
TH	thermal-hydrologic
THC	thermal-hydrologic-chemical
TSPA	total system performance assessment
UZ	unsaturated zone
WP	waste package
YMP	Yucca Mountain Site Characterization Project

TITLES OF KEY TECHNICAL ISSUES

Titles of Key Technical Issues (KTIs) are designated in text by acronym-like expressions and are placed within square brackets (e.g., [CLST]). The alphanumeric designation following the expression reflects the corresponding subissue or acceptance criterion in the issue resolution status report (IRSR) for that KTI (e.g., [CLST 5]). Following is a list of KTI expressions and their titles.

[CLST]	Container Life and Source Term
[ENFE]	Evolution of the Near-Field Environment
[IA]	Igneous Activity
[RDTME]	Repository Design and Thermal-Mechanical Effects
[RT]	Radionuclide Transport
[SD&S]	Structural Deformation and Seismicity
[TEF]	Thermal Effects on Flow
[TSPAI]	Total System Performance Assessment and Integration
[USFIC]	Unsaturated and Saturated Flow under Isothermic Conditions

INTRODUCTION TO THE TECHNICAL GUIDANCE DOCUMENT FOR LICENSE APPLICATION PREPARATION

This introduction provides information that applies to all chapters of the *Technical Guidance Document for License Application Preparation*. Authors preparing the License Application (LA) shall use this document for guidance and must read this Introduction and Appendix B before writing their respective sections.

This technical guidance document (TGD) was developed by the Civilian Radioactive Waste Management System (CRWMS) Management and Operating Contractor (M&O) for the U.S. Department of Energy (DOE). The document was prepared by an M&O team led by the M&O Licensing group.

This document was prepared using the proposed 10 CFR 63 (64 FR 8640) as the rule governing the granting of a license for the DOE to construct a potential repository at Yucca Mountain, and to receive and possess source, special nuclear, and by-product materials at the potential geologic repository. The DOE comments submitted to the U.S. Nuclear Regulatory Commission (NRC) on the proposed 10 CFR 63 (64 FR 8640), as contained in the "Interim Guidance Pending Issuance of New Nuclear Regulatory Commission (NRC) Regulations for Yucca Mountain, Nevada" (Dyer and Horton 1999), have been incorporated into this TGD.

PURPOSE AND SCOPE

This TGD provides guidance to authors for use in preparing the LA for a geologic repository for high-level radioactive waste (HLW). As identified in proposed 10 CFR 63.21(a) (64 FR 8640), the LA will consist of general information in Chapter 1 and a Safety Analysis Report in Chapters 2 through 13. The TGD and the LA have been evaluated using applicable approved procedures and determined to be non-Q documents. However, the LA will include and reference data that have been qualified in accordance with the Office of Civilian Radioactive Waste Management (OCRWM) Quality Assurance program.

Construction Authorization Compared with License to Receive and Possess Radioactive Waste–The TGD provides direction for the development of the LA to be submitted to the NRC. If, after review, the LA is approved by the NRC, then the NRC could authorize the construction of a repository at Yucca Mountain. The requirements for the issuance of, and conditions placed on, a construction authorization (CA) are defined in proposed 10 CFR 63.31 and proposed 10 CFR 63.32 (64 FR 8640), respectively. As defined in proposed 10 CFR 63.32(d) (64 FR 8640), a CA will be subject to the limitation that a license to receive and possess HLW will not be issued until the DOE has updated the LA, as specified in proposed 10 CFR 63.24 (64 FR 8640), and has met the requirements stated in proposed 10 CFR 63.41 (64 FR 8640). In the TGD, information to be provided in the LA for issuance of a CA will be differentiated from that required for issuance of a license to receive and possess HLW. The differentiated approach was adopted because some of the information needed to receive and possess the HLW will not be available at the time of docketing the LA, and because some information needed at the time of licensing is not needed to support the CA.

There are two situations that could occur relative to the information required at the time of docketing the LA (i.e., at CA) and the information required at the time of updating the LA to receive and possess HLW. In the first situation, all of the information needed at the time to receive and possess HLW is available at the time of docketing the LA. In this case, the guidance in the TGD will specify that the information be provided in the LA at the time of docketing (i.e., no differentiation in information to be provided in the LA). For such chapters, the information provided at the time of docketing will be brought up to date to reflect the changes, during construction, as of the LA update to receive and possess HLW. Chapters 3 and 7 are in this category.

In the second situation, some of the information needed at the time to receive and possess HLW will not be available at the time of docketing the LA, nor will it be needed for the NRC to issue a CA. In this case, the guidance in the TGD chapters will differentiate between the information required at the time of docketing (CA) and the information required at the time of LA update to receive and possess HLW. Specific guidance for providing information under this differentiated approach, consistent with proposed 10 CFR 63.24(b) (64 FR 8640), is delineated in each chapter, as appropriate.

Regulations–A cross-reference between the proposed 10 CFR 63 (64 FR 8640) regulations and the TGD sections that provide guidance related to those regulations is delineated in Appendix A.

The DOE has issued an interim guidance (Dyer and Horton 1999) to address project concerns with the proposed 10 CFR 63 (64 FR 8640). Appendix A also provides a cross-reference between the applicable sections of the interim guidance (Dyer and Horton 1999) and the TGD sections where that guidance is addressed. When the NRC publishes a final 10 CFR 63, references to the interim guidance will be removed and the final 10 CFR 63 will be incorporated into the TGD, as appropriate.

Guidance–Each chapter of the TGD includes guidance for demonstrating compliance with specific acceptance criteria and regulations. In addition, the TGD provides guidance for development of an appropriate safety case in the LA. The TGD also contains or refers to information, such as acceptable methods for analysis and testing, which may be used to demonstrate how the criteria are met. Developing an LA that adheres to this guidance should support successful docketing and eventual approval of the LA by the NRC. This introduction explains how language in the TGD is used to distinguish mandatory guidance from discretionary guidance.

In addition, a writer's guide for authors of the LA is provided in Appendix B. To ensure that the LA is written consistent with the *Style Manual for the Civilian Radioactive Waste Management System Management and Operating Contractor* (CRWMS M&O 1999), each LA author must read the writer's guide in Appendix B before beginning to write the LA.

THE TECHNICAL GUIDANCE DOCUMENT FORMAT

The TGD chapter and section titles are, in most cases, identical to those planned for the LA. The basic organization of each TGD chapter is identical to that of the other chapters. Each TGD chapter begins with a brief introduction to the subject matter addressed in that chapter. The rest

of the chapter provides guidance on material to be addressed under each section within the chapter. For the TGD, each chapter ends with a reference section that lists all of the references cited in that chapter of the TGD. For the LA, however, the last section in each chapter will list the references cited within that chapter of the LA, as opposed to the references in the corresponding chapter of the TGD.

GUIDANCE USAGE

Each chapter in the TGD corresponds to a chapter in the LA. Guidance in each chapter indicates whether a given item of guidance is mandatory. For example, an item stated as a command (e.g., "State that the design complies with Regulatory Guide 8.8," "Describe the system") is mandatory. Use of the word "should" denotes non-mandatory guidance. All authors must comply with requirements in this document, unless authorized in writing and approved by the M&O LA Development Manager. An electronic form for requesting and approving such deviations has been developed and is available in a Lotus Notes database.

Guidance in this TGD includes acceptance criteria deemed necessary for developing and documenting a safety case. Acceptance criteria are statements of compliance with regulations, codes, standards, etc. An acceptance criterion may read: "Describe the evaluation of the compliance of the waste package and its components with the structural design criteria in support of demonstrating compliance with proposed 10 CFR 63.112(e) (64 FR 8640)." Acceptance criteria within a chapter may also be developed from other sources. For example, the DOE may choose to invoke a requirement from an NRC regulation (e.g., 10 CFR 50) that, while not directly applicable to the repository, is considered appropriate for use as an acceptance criterion for the repository. Such a situation might occur for a regulation or guidance document that the DOE deems appropriate and has been used in licensing other facilities regulated by the NRC.

Identification of computer models and programs is specified in various locations in this TGD. It is not possible, however, to insert direction in every location in the TGD where a code or model might be discussed in the corresponding LA section. Additionally, other documents may be developed (e.g., topical reports) that have received approval by the NRC for the use of specific computer code or models. In general, if computer codes or models are used, then they must be identified and enough descriptive information provided to justify their use. If the code has been developed within the Yucca Mountain Site Characterization Project (YMP), then verification and validation consistent with the YMP Quality Assurance requirements must be provided. If a code that was not developed within the YMP is used, then a brief justification for use of that code must be provided. This justification may be a software or code verification, consistent with YMP procedures if the code was not approved by the NRC.

It is recognized that modifications to the information included in this TGD may be necessary for the LA, as determined by chapter lead authors or the technical leads. The TGD, as an approved YMP document, will inevitably lag behind the issues and information that potentially could affect licensing and the LA. Therefore, while writing the LA, lead authors and technical leads may deviate from the information required by this TGD, but only with the approval of the M&O LA Development Manager.

In general, information required by this TGD may be presented directly in the LA, or it may be included in the LA through reference to other documents. When other documents are referenced, sufficient detail about the data, analyses, or both, must be presented in the LA so that reviewers can understand the technical basis for any inference or conclusion drawn from the reference; the LA must identify those references. In some cases, the LA itself must contain certain information; in these cases, this TGD states that the individual section, not its references, must contain the information. Additional guidance for use of references is provided in the *Management Plan for the Development of the License Application for a High-Level Waste Repository at Yucca Mountain* (YMP 1999). In all cases, references and data used to support the LA must be traceable to the Technical Data Management System, to the Records Information System, or to one of the YMP technical information centers as required by project procedures.

In some instances, this TGD provides directions to identify the guidance documents to which a given process complies or will comply with (e.g., regulatory guides, industry codes, and standards). Typical practice in the commercial nuclear industry is to state that a process complies with a specific revision of a guidance document. If the guidance document is revised or updated, the licensee is not automatically required to commit to, or comply with, the updated guidance. Therefore, when this TGD requires a statement of compliance with a regulatory document, the authors must determine the appropriate revision of the guidance with which compliance is sought. The authors must then explicitly identify, in the LA, the revision to which the DOE commits to comply.

Revisions to the TGD are expected to address changes in available guidance, changes to project compliance strategies, changes to applicable regulations, and other eventualities. This TGD, and the draft LA, will be revised as appropriate to reflect these changes. In addition, this TGD also may be revised at the direction of the Yucca Mountain Site Characterization Office Assistant Manager for Licensing and Regulatory Compliance.

LEVEL OF DETAIL

In general, the level of detail for information in the LA is the level needed to present the safety case (i.e., demonstrate compliance with guidance and acceptance criteria provided in the TGD) with minimal need to consult references. In other words, the required level of detail is that needed to demonstrate compliance with regulations. Demonstrating compliance involves providing sufficient technical basis to allow the NRC to determine that there is reasonable assurance that the repository can and will be designed, constructed, and operated without unreasonable risk to the health and safety of the public, and to demonstrate that there is reasonable assurance that postclosure performance objectives will be met, consistent with the regulations. Providing unnecessary detail for these demonstrations should be avoided, although references may be used to point to additional detail in a given area. Ultimately, the level of detail that will be provided in the LA is defined in this TGD, possibly augmented by other YMP documents, and refined in the development of the LA and NRC review process.

It should be recognized that, while a particular topic probably should be presented at a substantial level of detail, there may be information on that topic that is missing from the LA discussion. The NRC has explicitly recognized the potential for incomplete information at the time of the LA submittal. In Section 11.13 of the LA, structures, systems, and components that

require research and development to confirm the adequacy of design will be summarized, a plan for obtaining the needed information will be presented, and schedules for obtaining the information will be provided.

SPECIFIC ISSUES AND AREAS OF INTEREST

Consistency with Environmental Impact Statement and Site Recommendation Report–Authors of the LA must ensure that data and information presented in the LA are consistent with the data and information on corresponding topics presented in the repository environmental impact statement (EIS), which is to accompany the LA and other reports, such as the Site Recommendation Report (SRR). Each author of the LA must review these documents for consistency with the LA. Because the repository design will continue to evolve after publication of the EIS and the SRR, there will be some differences between these documents and the LA. Authors of the LA must be aware of these differences and must inform management, as appropriate, of changes that potentially invalidate the information presented in the EIS and the SRR.

Potentially Applicable Regulatory and Industry Guidance–In the LA, it is important to explicitly identify the NRC regulatory guides, industry codes and standards, and other applicable documents with which the DOE will comply. For these documents, authors must include version numbers, revision numbers, or dates of issuance.

The DOE will comply with the applicable regulatory guides that have been issued by the NRC, with some possible exceptions. The DOE may choose to comply with other regulatory guides not directly applicable to a geologic repository if compliance with the other regulatory guides is believed to be appropriate, or if doing so will facilitate licensing. The LA will include tables in Section 2.3 that summarize regulatory guidance documents (including revision numbers) and the extent (fully or partially) to which the YMP will comply with each guidance document. Also in Section 2.3 of the LA, reference will be made to other sections of the LA where these topics are discussed further.

In addition to regulatory guides, various other forms of regulatory and industry guidance are available for use in the LA. These other documents include NUREGs (technical documents written by or for the NRC) and industry codes and standards (e.g., the ASME Boiler and Pressure Vessel Code). Where determined to be appropriate, the TGD provides guidance to authors of the LA on how to discuss the manner in which the YMP uses these documents. Generally, however, the TGD refers the LA author to source documents that state the YMP position on the use of specific documents.

U.S. Nuclear Regulatory Commission Key Technical Issues–The NRC has issued nine key technical issues (KTIs) that it believes must be resolved before the repository can be licensed. These issues have served as bases for NRC priorities in the HLW disposal and, ultimately, they will be incorporated in the NRC Yucca Mountain review plan.

The NRC has provided issue resolution status reports (IRSRs) that document its opinion on the status of resolution of corresponding KTIs and further subdivide the KTIs into subissues. The NRC has identified various acceptance criteria that will be used by NRC staff to review the LA.

Because of the focus by the NRC and the DOE on the KTIs, the TGD specifies how each subissue and acceptance criterion in the IRSRs will be addressed in the LA.

In the TGD, the guidance for discussions of the KTIs in the LA is not contained in one complete section but is distributed throughout the document where applicable subjects are discussed. Generally, the compliance discussions incorporate the acceptance criteria and subissues in a continuous manner to the extent possible. Special headings have been used, as needed, to start discussions of an acceptance criterion if the discussion flow requires it. Guidance in the TGD regarding acceptance criteria mirror the actual statements of acceptance criteria to the extent possible. For example, if an acceptance criterion states that adequate data must be available for a particular item, then the TGD will specify that authors of the LA are to demonstrate that adequate data is available on that particular item. Simple parenthetical references are included to identify, in the TGD, discussions specifically related to IRSR acceptance criteria and subissues.

Some exceptions to including every IRSR acceptance criterion and subissue in the TGD have been taken. Excluded KTIs include those covering areas such as programmatic issues, which normally are covered external to an LA; technical issues that are logically covered elsewhere; and subissues or acceptance criteria to which the YMP has taken, or is expected to take, exception.

Full compliance with applicable IRSR acceptance criteria is, in many cases, associated with the time of update of the LA to receive and possess HLW. Complete compliance with the IRSR acceptance criteria may not be possible for some KTIs at the time of docketing the LA. There are plans to establish compliance thresholds for IRSR acceptance criteria that cannot be fully resolved at the time of docketing the LA. These plans entail interactions between the NRC and the DOE. For these IRSR acceptance criteria, information that demonstrates compliance to the established threshold will be provided in the LA at the time of docketing (for CA). Consistent with the differentiated approach, discussed earlier, full compliance with applicable IRSR acceptance criteria will be demonstrated with the update to LA to receive and possess HLW.

The status of each IRSR acceptance criterion and subissue is tracked (whether or not it is included in the TGD) in a Lotus Notes database maintained by the M&O Licensing group. This database also tracks exceptions taken, by the Project, to the acceptance criteria.

Proprietary, Privileged, or Private Information–The DOE will attempt to maximize the availability of pertinent licensing information to the stakeholders. It is expected that little proprietary or privileged information, or information addressed by the Privacy Act of 1974, will be needed to support the LA. If proprietary information needs to be submitted to support the LA, then proprietary information must be submitted separately, must be identified clearly, and must be accompanied with detailed reasons for requesting withholding of the document from public disclosure (as specified by Section 2.790 of 10 CFR 2, Rules of Practice for Domestic Licensing Proceedings and Issuance of Orders). Privileged information, or information protected by the Privacy Act of 1974, will not be included in the LA; this information will be available for regulatory authorities to examine as required. Other agencies wishing to access this information must request it in accordance with the Privacy Act of 1974.

Classified Information–Classified government information may be necessary to support submittal of the LA. As specified in Section 2.790 of 10 CFR 2, classified information may be withheld from public disclosure if it is covered by an appropriate executive order. Classified information should be submitted as a separate document in accordance with the information security program of the affected agency.

Unclassified Sensitive Information–Unclassified sensitive information will be necessary to support submittal of the LA. As specified in Section 2.790 of 10 CFR 2, unclassified sensitive information (i.e., safeguards and security information) must be withheld from public disclosure. Such information will be submitted separately from the LA.

STATE OF COMPLETION OF THE TECHNICAL GUIDANCE DOCUMENT AND PLANS FOR FUTURE REVISIONS

This TGD has been prepared as an assessment of regulatory and industry guidance applicable to development of an LA for a potential repository. Revisions are anticipated for various reasons, including:

- The NRC plans to issue final 10 CFR 63 regulations in early 2000. This will require changing the TGD guidance to address the final regulations.
- Additional guidance may be provided to support the update of the LA to receive and possess HLW. This document will be revised to include additional guidance.
- It is possible that the repository design will change enough to make some guidance inapplicable. This document will be revised to exclude inapplicable information.
- As preparation of the LA proceeds, additional technical issues may emerge that will call for revisions. This document will be revised to include new information.
- Regulatory documents, industry guidance, and regulations that serve as sources of guidance in this TGD may change during development of the LA. This document will be revised to reflect these changes.
- The DOE will need to commit to specific revisions of given guidance documents (as specified in the LA), but the LA will not be updated automatically when the guidance documents are revised. However, the date for making decisions about committing to specific revisions has not been determined. In the interim, this TGD may be revised, if necessary, to reflect the latest revisions of the regulatory guidance documents before a decision to commit to a particular revision is made.

Revisions to the TGD are anticipated, so to the extent that the existing draft text for the LA is affected by these revisions, lead authors will need to revise text to appropriately incorporate the revised content guidance.

REFERENCES

References Cited

CRWMS (Civilian Radioactive Waste Management System) M&O (Management and Operating Contractor) 1999. *Style Manual for the Civilian Radioactive Waste Management System Management and Operating Contractor.* B00000000-01717-3500-00004 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990824.0240.

Dyer, J.R. and Horton, D.G. 1999. "Interim Guidance Pending Issuance of New Nuclear Regulatory Commission (NRC) Regulations for Yucca Mountain, Nevada." Letter from J.R. Dyer (DOE/YMSCO) to D.R. Wilkins (CRWMS M&O), June 18, 1999, OL&RC:AVG:1435, with enclosure, "Interim Guidance Pending Issuance of New NRC Regulations for Yucca Mountain." ACC: MOL.19990712.0039.

YMP (Yucca Mountain Site Characterization Project) 1999. *Management Plan for the Development of the License Application for a High-Level Waste Repository at Yucca Mountain.* YMP/97-02, Rev. 1. Las Vegas, Nevada: Yucca Mountain Site Characterization Office. ACC: MOL.19990624.0242.

Codes, Standards, and Regulations

10 CFR (Code of Federal Regulations) 2. Energy: Rules of Practice for Domestic Licensing Proceedings and Issuance of Orders. Readily available.

10 CFR 50. Energy: Domestic Licensing of Production and Utilization Facilities. Readily available.

64 FR (Federal Register) 8640. Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, Nevada. Proposed rule 10 CFR 63. Readily available.

Privacy Act of 1974. 5 U.S.C. 552a et seq. Readily available.

CHAPTER 1. INTRODUCTION AND GENERAL INFORMATION

State that this chapter provides a high-level introduction to the License Application (LA). State that the discussions presented here provide the general information portion of the LA required by Interim Guidance Section 21(a) (Dyer and Horton 1999) and proposed 10 CFR 63.21(b) (64 FR 8640), information required by other proposed 10 CFR 63 (64 FR 8640) requirements, and an overview of the information presented in the Safety Analysis Report (SAR) portion (Chapters 2 through 13) of this LA. The authors of Chapter 1 will produce a chapter that allows the reader to gain a general understanding of the potential monitored geologic repository (MGR) facility, the surrounding environment, schedule, operations, programs, and performance. References to detailed discussions of material in the SAR chapters also will be provided in this chapter. Authors preparing the LA shall use this document for guidance and must read the Introduction and Appendix B before writing their respective sections.

The guidance presented in this chapter applies for docketing of the LA at the time of construction authorization (CA) and at the time of updating the LA to receive and possess high-level radioactive waste (HLW). All of the information delineated in this chapter must be presented in the LA at the time of docketing the LA. The information presented in the LA at the time of CA will be confirmed to be correct or brought up-to-date for the LA to receive and possess HLW.

1. INTRODUCTION

Information Required at the Time of Construction Authorization–State that the purpose of this section is to introduce the purpose of the LA, its layout, and contents. Additionally, this section should provide a brief background of the site selection process, introduce the potential MGR, and notify the reader of the availability of all documentary material providing a basis for the safety cases as presented in this LA.

State that the purpose of the LA submittal to the U.S. Nuclear Regulatory Commission (NRC) is to support issuance of a CA. State that the LA consists of the general information in this chapter and a SAR consisting of Chapters 2 through 13. Point out that the environmental impact statement (EIS) that complies with the Nuclear Waste Policy Act of 1982, as amended (NWPA), will accompany the LA to satisfy the requirements of proposed 10 CFR 63.22(a) (64 FR 8640).

State that the LA reflects a differentiated approach between LA for CA and LA to receive and possess HLW, as specified in proposed 10 CFR 63.24 (64 FR 8640). State that the LA satisfies the requirements and conditions as defined in proposed 10 CFR 63.31 and proposed 10 CFR 63.32 (64 FR 8640), respectively, for a CA. State that the differentiated approach was adopted because some of the information necessary at submittal of the LA to receive and possess is unavailable or unneeded to support the CA.

State that the term "Monitored Geologic Repository," used throughout the LA, is equivalent in meaning to "repository," as defined and used in the NWPA. Continue the introduction by providing a brief description of the regulatory bases and requirements from the applicable NRC regulations. Provide a functional description of the objectives of the potential MGR. Provide its geographic location.

Describe the scope of the contents of the LA, including each chapter, appendix, and those portions submitted in separate documents that are withheld from public disclosure. Cite 10 CFR 2.790 for the authority to withhold.

State that, pursuant to the requirements in 10 CFR 2, Subpart J, this LA and any other documentary material is being provided electronically for placement in the NRC licensing support network. In 10 CFR 2, documentary material means the information upon which the U.S. Department of Energy (DOE) bases its positions, the information relevant to but not supporting the DOE positions, and the reports and studies relevant to the LA and topical guideline issues in Regulatory Guide 3.69, *Topical Guidelines for the Licensing Support System* (NRC 1996a). The scope of documentary material is to be determined by the topical guidelines in the applicable NRC regulatory guide (10 CFR 2.1001, Definitions, Documentary Material).

Site Selection–Briefly relate the historical background pertinent to the Yucca Mountain site selection. Note the statutory direction regarding the study of Yucca Mountain only. Provide an overview of the geographic setting of the site, and note significant natural features that contribute to the protection of the public health and safety. Provide a map showing the location of Yucca Mountain in Nevada.

Information Required at the Time of Update to the License Application to Receive and Possess–State the purpose of the LA submittal to the NRC is to support issuance of a license to receive and possess HLW. State that the EIS submitted with the LA for CA has been updated as appropriate to support NRC granting a license to receive and possess in accordance with proposed 10 CFR 63.41(e) (64 FR 8640). For the license to receive and possess HLW, the initial term requested will be 40 years from the date of issuance.

1.1 GENERAL DESCRIPTION OF THE REPOSITORY

State that the purpose of this section is to satisfy the requirements of the proposed 10 CFR 63.21(b)(1) (64 FR 8640). Proposed 10 CFR 63.21(b)(1) (64 FR 8640) requires a general description of the potential MGR at the Yucca Mountain site, identifying the location of the potential MGR, the general character of the proposed activities, and the basis for the exercise of the licensing authority of the NRC.

To satisfy this requirement, the author is to provide a high-level overview of the potential MGR location and surrounding environment, layout and description, major activities and operational concepts, description of radioactive materials to be received during operation, and the basis for the licensing authority. Detailed information in Chapters 3, 4, 5, 6, and 11 is available for the high-level overviews of all subsections except Subsection 1.1.4. Provide a reference to the appropriate chapter.

The general description presented in the following sections provides primarily an executive summary of information presented elsewhere in the LA (NRC 1996b, Section 1.4.2).

1.1.1 Location and Description of the Repository

State that this subsection presents a high-level description of the areas surrounding the Yucca Mountain site, the natural systems of the geologic setting, and the potential MGR facility. Demonstrate compliance with the requirements stated in proposed 10 CFR 63.21(b)(1) (64 FR 8640). Provide a general summary-level discussion of the topics presented below, and refer to more detailed discussions of these topics in later chapters and sections.

Describe, at a top level, the repository site, its location, including major cities and populations in the vicinity, and the geologic setting. In this description, summarize outstanding geologic, hydrologic, meteorologic or climatologic, geochemical, geographic, and demographic features of the site as well as nearby industrial, transportation, and military facilities. Provide a reference to Chapter 3 for a more complete discussion.

Describe in general terms, using illustrations as appropriate, the surface and underground layouts of the potential MGR. Briefly describe the activities to be performed in each surface structure (e.g., the security building, Carrier Preparation Building, Waste Handling Building, Waste Treatment Building, North Portal, and South Portal). Provide a general overview of the functions of the systems for handling spent nuclear fuel (SNF) and HLW. Note the key design features of the underground facility (e.g., depth of the underground repository, dimensions and locations of waste emplacement tunnels and panels, placements of waste packages [WPs], and distance between WPs).

Describe, at a high level, the geologic repository preclosure and postclosure controlled area boundaries, radiologically controlled area boundary, operations area, major site structures, Quality Level 1 and 2 structures, systems, and components (SSCs), and engineered barriers on the site. Boundaries, both natural and man-made, should be discussed briefly and depicted on simple drawings or maps. Provide a reference to Chapters 4, 5, and 6 where the potential MGR and accessible environment boundaries are detailed.

Reference the chapters of the LA that contain more detailed design and operation information about the underground facility.

1.1.2 Description of Radioactive Materials to Be Received

State that this subsection summarizes the kind, amount, and specifications of the radioactive material proposed to be received and possessed at an MGR at the Yucca Mountain site. Proposed 10 CFR 63.21(c)(12) (64 FR 8640) requires a description of the kind, amount, and specifications of the radioactive material proposed to be received and possessed at an MGR. Provide a reference to the information in Chapter 5 (i.e., a description of the kinds, amounts, and specifications of the radioactive materials to be received and possessed for which a CA or license to receive and possess is being pursued). Provide a description of the design basis contents of the WPs for such wastes. In addition, provide the total quantity of radioactive material that the potential MGR is designed to handle, in consideration of the statutory limits. Note the modes of transport of such materials to the site.

1.1.3 General Description of Proposed Activities

State that this subsection describes, at a high level, the expected operations at the potential MGR. Demonstrate compliance with the description of activities stated in proposed 10 CFR 63.21(b)(1) (64 FR 8640). Provide a brief discussion of the concept of operations during all phases of the project (i.e., pre-emplacement construction, emplacement operations, monitoring, closure, and decommissioning). Briefly discuss plans for maintaining the waste retrieval option. Refer to the additional discussion of these topics in Section 11.12. Also provide a reference to Section 1.2 for the planned schedule for each phase of the repository project.

Provide an integrated overview of the waste-handling concepts planned for use. This can be accomplished by following the path of fuel from arrival at the site and inspection at security through MGR closure.

Include in the overview appropriate illustrations for the basic waste-handling operations concepts for the repository. This discussion will integrate the individual SSC operating concepts described in Chapter 4 to show how the repository operating crews will control major repository equipment to perform the waste-handling operations of waste receipt and surface handling and Chapter 6 for emplacement.

Include in the overview brief discussions of the significant design features related to waste-handling operations and low-level radioactive waste generation used to ensure radiological safety for the public and the repository workers. Affirm that as low as is reasonably achievable (ALARA) principles have been incorporated into the design features and concept of operations for the facility. Address subjects such as use of transfer cells and spent fuel pools, waste management systems, ventilation confinement, and the features provided to control worker radiation exposures and to limit the spread of contamination during normal and design basis events. Provide a reference to Chapters 2, 4, 7, 9, 10, and 11, as appropriate, in which these features are described in more detail.

1.1.4 Basis of Licensing Authority

State that this subsection presents the basis of the licensing authority for disposal of HLW and SNF in the potential MGR and the respective roles of the NRC, U.S. Environmental Protection Agency, and the DOE. Demonstrate compliance with the requirement stated in proposed 10 CFR 63.21(b)(1) (64 FR 8640). State that, pursuant to the NWPA, Section 8(c), the DOE, as an applicant for a license to construct and operate an MGR, is subject to federal law and NRC regulations applicable to siting and construction, as well as the transfer, possession, and disposal of HLW. Include a discussion of the role of the U.S. Environmental Protection Agency in setting the applicable environmental performance standard for the repository for radioactive releases to the environment. Refer to the applicable sections of the Atomic Energy Act of 1954, as amended, and Section 202 of the Energy Reorganization Act of 1974. Also refer to Title I, Subtitle A, of the NWPA, which discusses the establishment of a site characterization program, site selection, and other activities relating to repositories for the disposal of HLW and SNF. Provide a chronological history of how the responsibility of final disposition of HLW was assigned to the DOE, and how the NRC received licensing authority for HLW. Reference all

appropriate provisions of the statutory authority to perform the various activities at the repository and NRC regulations applicable to the proposed activities at the potential MGR.

1.2 PROPOSED SCHEDULES FOR CONSTRUCTION, WASTE RECEIPT, AND WASTE EMPLACEMENT

State that the purpose of this section is to present a complete overview of the scheduled activities for the phases of the potential MGR, except for the initial licensing phase.

Demonstrate compliance with the requirements stated in the proposed 10 CFR 63.21(b)(2) (64 FR 8640) by providing schedules, from start of construction through closure. The schedules must include construction, receipt of waste, waste emplacement, and closure as required by the regulation. Ensure the schedules used are the approved schedules for the Office of Civilian Radioactive Waste Management program. Discuss assumptions integrated into the schedules as necessary.

1.3 IDENTIFICATION OF AGENTS AND CONTRACTORS

State that this section identifies to the NRC the key positions in construction and operation organizations responsible for the safety and operation of the potential MGR. This section should demonstrate with reasonable assurance that the individuals in key positions and primary contracted organizations are capable of constructing and operating a facility safely and protecting the public, workers, and the environment.

Proposed 10 CFR 63.21(c)(22)(i) (64 FR 8640) requires discussion of the organizational structure of the DOE as it pertains to construction and operation of the MGR. This includes a description of any delegations of authority and assignments of responsibilities, whether in the form of regulations, administrative directives, contract provisions, or otherwise. Also, identification of key positions that are assigned responsibility for safety at and operation of the MGR is required by proposed 10 CFR 63.21(c)(22)(ii) (64 FR 8640).

State that more information is available in Section 11.1 but provide a brief overview of the following information. Indicate that the DOE is the applicant, has responsibility for the MGR, and has contracted or will contract with the appropriate contractor(s) for the design, analysis, construction, inspection, and operation of the facility. If the information is available, identify the prime agents or contractors involved in the design and construction of the repository and its associated SSCs by name, including any management and operating contractors and all team member companies. Also, provide the prime agents and contractors that will be (or are expected to be) involved in operating the facilities, including any delegations of authority and responsibility. It is not necessary to provide the addresses and identification of principal contacts of agents and contractors. Identify all principal consultants and outside service organizations by name, including those providing quality assurance (QA) services. Provide the professional qualifications of agents and contractors. Describe the division of responsibility among the designer, architect-engineer, and constructor (NRC 1989, Section 1.4).

Construction and operation activities must be conducted under a QA program approved by the NRC, as required by DOE QA procedures. State that all contractors will work under either the

DOE NRC-approved QA program or their NRC-approved programs (NRC 1996b, Section 1.4.4).

Where prime agents and contractors have not been identified, state that all prime agents and contractors will be qualified for their scopes of work and will follow the DOE QA program approved by the NRC or their own NRC-approved QA program. State that responsibilities of these agents and contractors will be specified.

1.4 DESCRIPTION OF SITE CHARACTERIZATION AND PERFORMANCE ASSESSMENT WORK CONDUCTED

State that this section provides high-level summaries of the site characterization work conducted for the Yucca Mountain site and the transition to performance confirmation activities. State that the section also presents a high-level summary of the results from the integrated safety analysis and total system performance assessment for preclosure and postclosure performance, respectively.

1.4.1 Summary of Site Characterization Activities

State that the purposes of this subsection are to present the site characterization work conducted for the Yucca Mountain site and to discuss, at a high level, the transition of the site characterization program work scope from that originally planned to that conducted. Provide a high-level overview of the work conducted for site characterization at the program level to satisfy the requirements of proposed 10 CFR 63.21(b)(5) (64 FR 8640). Identify the major features, events, and processes likely to influence the design and performance of the potential MGR. Focus on the results or conclusions drawn regarding the performance of the natural systems of the geologic setting in tandem with the engineered portions of the potential MGR.

Briefly introduce the *Site Characterization Plan: Yucca Mountain Site, Nevada Research and Development Area* (DOE 1988) and its broad investigation coverage to ensure natural systems, design, and performance uncertainties would be resolved. State that the originally proposed site characterization activities were based on limited knowledge of the site features, events, and processes and a differing conceptual design. Explain that as more understanding of the site was gained, a different conceptual design was chosen, performance assessment tools matured, and the waste program's needs changed, the site characterization program was modified to its current form.

Discuss at the program level (i.e., the 11 primary divisions of the site characterization program) the changes in work scopes since completion of the *Site Characterization Plan: Yucca Mountain Site, Nevada Research and Development Area* (DOE 1988) and the actual work conducted. Use *Documentation of Program Change* (CRWMS M&O 1999) as the source for the history of the site characterization program; the document links activities conducted during the site characterization phase to the proposed activities described in the 1988 site characterization plan. Reference the revision current at time of LA submittal. Inform the reader of the document's role as providing an historical perspective of the site characterization program at Yucca Mountain and that the document is not a quality-assured record. Avoid detailed comparisons between the

originally proposed site characterization program and the implemented program in the discussion.

Describe the transition from site characterization to performance confirmation and address continuing testing activities.

State that the site characterization program as conducted has yielded sufficient, qualified, and appropriate data to determine with reasonable assurance that the performance of the natural systems of the geologic setting and engineered portions of the potential MGR will not endanger public, worker, or environmental safety during the preclosure or postclosure periods.

1.4.2 Statement of Compliance and Performance Assessment Summary

State that this subsection presents the calculated doses for the preclosure operational and postclosure periods. State that the MGR, WP, engineered barrier system designs, and natural systems of the geologic setting provide assurance that the preclosure performance objectives of proposed 10 CFR 63.111 (64 FR 8640) and the postclosure performance objectives of proposed 10 CFR 63.113 (64 FR 8640) will be satisfied. Include in this section the expected annual dose rates for normal and off-normal operations and design basis events during preclosure operations.

Present the expected annual dose rates over the 10,000-year postclosure period of regulatory compliance. State that the preclosure integrated safety analysis and the postclosure performance assessments address how the repository meets the performance objectives related to radiological protection identified in Subpart E of proposed 10 CFR 63 (64 FR 8640).

State that Section 2.1 introduces the integrated safety analysis required by proposed 10 CFR 63.111(c) (64 FR 8640) and describes how that analysis is addressed in the LA. State that Section 2.1 also introduces the performance assessments required by proposed 10 CFR 63.113(c) and (d) (64 FR 8640) and describes how the assessments are addressed in the LA. State that Section 2.2 summarizes the comprehensive postclosure safety case for the potential MGR, including how containment and isolation of waste is accomplished through reliance on natural and engineered barriers.

1.5 QUALITY ASSURANCE

Proposed 10 CFR 63.21(c)(11) (64 FR 8640) requires that the LA present a description of the QA program applied to the SSCs important to safety and to the engineered and natural barriers important to waste isolation. Provide demonstration that the QA program, through planned and systematic processes, assures the MGR can be safely designed, constructed, operated, and closed; include a brief discussion on the aspect of quality control in the QA program (proposed 10 CFR 63.141 [64 FR 8640]).

Introduce this section by stating that the Office of Civilian Radioactive Waste Management QA program applied to the SSCs important to safety and to the engineered and natural barriers important to waste isolation, per proposed 10 CFR 63.142 (64 FR 8640), is embodied in the current *Quality Assurance Requirements and Description* (QARD) (DOE 1998) document. State that each QARD (DOE 1998) revision after Revision 0 has been reviewed and accepted by NRC

and, therefore, this section need not restate the currently accepted Office of Civilian Radioactive Waste Management (OCRWM) QA program.

Briefly describe how the QARD (DOE 1998) is based on the criteria of 10 CFR 50, Appendix B, and supplemented by other applicable Code of Federal Regulations (CFR), NRC guidance, and nuclear industry standards. Describe how the QARD (DOE 1998) relates to the OCRWM organizational structure and how affected organizations translate applicable QARD (DOE 1998) requirements into implementing documents for their scope of work. Describe how a matrix of implementing documents is developed, documented, and maintained to ensure QA program requirements are appropriately applied to work processes (proposed 10 CFR 63.143 [64 FR 8640]).

State that the OCRWM QA program is applied to all SSCs important to safety and to design and characterization of barriers important to waste isolation, as classified in accordance with QARD (DOE 1998, Section 2.2.2). State that the QA program is also applied to control activities as specified in QARD (DOE 1998, Section 2.2.3). State that application of the QA program is graded in accordance with QARD (DOE 1998, Section 2.2.4). Provide a reference to Section 2.4 of this LA for a description of the classification and grading processes applied to SSCs.

State that implementation of the QA program is continually verified through independent QA audits and surveillance and by management assessments conducted by personnel outside the QA organization, in accordance with the QARD (DOE 1998).

State that the current OCRWM QA program will be applied to SSCs and the related activities at the Yucca Mountain repository site from design to closure and decommissioning.

1.6 PHYSICAL PROTECTION AND MATERIAL CONTROL AND ACCOUNTING

State that the purpose of this section is to identify and describe the safeguards and security plans necessary to protect the nuclear materials at the potential MGR. The physical protection and material control and accounting programs, their plans, and implementing procedures are to demonstrate that adequate controls are in place at the MGR to ensure activities will not be adverse to the common defense and security. This demonstration is required to satisfy proposed 10 CFR 63.31(b) (64 FR 8640) for obtaining CA. State that this section satisfies the requirements of Interim Guidance Section 21(b)(3) (Dyer and Horton 1999) and 10 CFR 63.21(b)(4) (64 FR 8640), to describe the plans for physical protection at the MGR and the material control and accounting program.

Introduce the four plans that have been identified to satisfy proposed 10 CFR 63.31(b) (64 FR 8640):

- Physical Protection Plan (Interim Guidance Section 21(b)(3) [Dyer and Horton 1999])
- Safeguards Contingency Plan (Interim Guidance Section 21(b)(3) [Dyer and Horton 1999])

- Security Organization Personnel Training and Qualification Plan (Interim Guidance Section 21(b)(3) [Dyer and Horton 1999])

- Material Control and Accounting Plan (proposed 10 CFR 63.21(b)(4) [64 FR 8640])

State that the plans contain safeguards information, or unclassified sensitive information, and are withheld from public disclosure as exempted under 10 CFR 2.790.

Discuss the purposes and scopes of the individual plans in a broad overview. Do not provide specific details that could weaken protection of the nuclear materials and safeguards. State that the Physical Protection Plan, the Safeguards Contingency Plan, and the Security Organization Personnel Training and Qualification Plan are based on the requirements of 10 CFR 73.51, other applicable requirements that result from 10 CFR 73.51, and Safeguards and Security compliance program guidance packages and system description documents (SDDs). Include discussion of tests, inspections, audits, and any other means that will demonstrate compliance with the 10 CFR 73.51 regulation.

State that the Material Control and Accounting Plan is based on the requirements of proposed 10 CFR 63.78 (64 FR 8640), other applicable requirements that result from proposed 10 CFR 63.78 (64 FR 8640), and, if available, compliance program guidance packages and SDDs for material control and accounting. State that the plan will ensure proper record keeping for pertinent material control and accounting information (proposed 10 CFR 63.71(a) [64 FR 8640]) and Interim Guidance Section 71(b) [Dyer and Horton 1999]). Provide a reference to Section 11.4 for more detailed information on records.

Ensure this section is in full agreement with the information contained in the individual plans developed.

For the Individual Plans–Proposed 10 CFR 63.31(b) (64 FR 8640) requires that there is reasonable assurance that the activities proposed in the application will not be inimical to the common defense and security.

Provide purpose and high-level description statements for the four plans (meeting the requirements of Interim Guidance Section 21(b)(3) [Dyer and Horton 1999] and proposed 10 CFR 63.21(b)(4) [64 FR 8640]). It should be noted that the program and implementation of the plans, and eventually working procedures, are to provide reasonable assurance for the common defense and security. Discuss how the requirements of the applicable regulations are satisfied.

Discuss how the Safeguards and Security plans meet the physical protection requirements in 10 CFR 73.51, including discussion of tests, inspections, audits, and any other means which demonstrate compliance. For the material control and accounting plan, discuss how proposed 10 CFR 63.78 (64 FR 8640) requirements are satisfied in that plan. Also include records requirements per proposed 10 CFR 63.71(a) (64 FR 8640) and Interim Guidance Section 71(b) (Dyer and Horton 1999). Ensure the applicable requirements, as identified in an applicable compliance program guidance package or Safeguards and Security SDD, are addressed in the appropriate plans.

1.7 MATERIALS INCORPORATED BY REFERENCE

State that, effective January 29, 1999, and pursuant to the requirements in 10 CFR 2, Subpart J, DOE has provided the documentary material electronically for placement in the NRC licensing support network in order to be docketed. Documents that are incorporated by reference will be available to the public. (NOTE: Previous hard-copy submittals do not meet this requirement and must be made available electronically.)

Demonstrate compliance with the referencing of related information incorporated into the LA as required by proposed 10 CFR 63.23 (64 FR 8640). For each document incorporated by reference, provide the title, the report number, the date submitted to the NRC, and the sections of the LA in which this report is referenced. If the report is used in numerous sections of the LA, state that it is referenced throughout.

- Provide a record of all topical reports previously submitted to the NRC, which are incorporated by reference as a part of the LA. This record includes reports that have been prepared by the DOE, its prime agents or contractors, or other organizations, and filed separately with the NRC in support of the LA.

- Results of tests and analyses may be submitted as separate reports. For these documents, provide the reference information including title, report number, date of issuance, and the section of the LA referenced. Also, include a summary in the appropriate section of the LA in which this information is used.

- For reports proposed to be withheld from public disclosure pursuant to 10 CFR 2.790(b) and 10 CFR 2.790(d) as proprietary documents or reports containing restricted data, national security information, or unclassified sensitive information, nonproprietary summary descriptions and unclassified summaries of the general content of such reports must be referenced.

- Also include a separate list of documents, which have been submitted to the NRC in other applications external to the Yucca Mountain Site Characterization Project, that are incorporated in whole or in part by reference in this LA (e.g., topical safety analysis reports and SARs for packaging submittals for storage, transportation, or WP systems for SNF or HLW that are to be used at the MGR).

The record does not include references cited in the LA but not submitted to the NRC. Because the final EIS is to accompany the submittal of the LA to the NRC, it shall not be incorporated by reference here. Also not included in the record are NRC regulations and regulatory guides.

1.8 REFERENCES

The following references were used to develop this chapter of the TGD. For the LA, this section will contain the references used to develop this chapter of the LA.

References Cited

CRWMS (Civilian Radioactive Waste Management System) M&O (Management and Operating Contractor) 1999. *Documentation of Program Change*. B00000000-01717-5700-00021 REV 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990712.0161.

DOE (U.S. Department of Energy) 1988. *Site Characterization Plan: Yucca Mountain Site, Nevada Research and Development Area*. DOE/RW-0199. Eight Volumes. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management, Nevada. ACC: HQO.19881201.0002.

DOE 1998. *Quality Assurance Requirements and Description*. DOE/RW-0333P, Rev. 8. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.19980601.0022.

Dyer, J.R. and Horton, D.G. 1999. "Interim Guidance Pending Issuance of New Nuclear Regulatory Commission (NRC) Regulations for Yucca Mountain, Nevada." Letter from J.R. Dyer (DOE/YMSCO) to D.R. Wilkins (CRWMS M&O), June 18, 1999, OL&RC:AVG:1435, with enclosure, "Interim Guidance Pending Issuance of New NRC Regulations for Yucca Mountain." ACC: MOL.19990712.0039.

NRC (U.S. Nuclear Regulatory Commission) 1989. *Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage)*. Regulatory Guide 3.48, Rev. 1. Washington, D.C.: U.S. Nuclear Regulatory Commission. Readily available.

NRC 1996a. *Topical Guidelines for the Licensing Support System*. Regulatory Guide 3.69. Washington, D.C.: U.S. Nuclear Regulatory Commission. Readily available.

NRC 1996b. *Standard Review Plan for Spent Fuel Dry Storage Facilities*. NUREG-1567. (Draft Report for Comment). Washington, D.C.: U.S. Nuclear Regulatory Commission. TIC: 226657.

Codes, Standards, and Regulations

10 CFR (Code of Federal Regulations) 2. Energy: Rules of Practice for Domestic Licensing Proceedings and Issuance of Orders. Readily available.

10 CFR 50. Energy: Domestic Licensing of Production and Utilization Facilities. Readily available.

10 CFR 73. Energy: Physical Protection of Plants and Materials. Readily available.

64 FR (Federal Register) 8640. Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, Nevada. Proposed rule 10 CFR 63. Readily available.

Atomic Energy Act of 1954. 42 U.S.C. 2011 et seq. Readily available.

Energy Reorganization Act of 1974. Public Law 93-438, 88 Statute 1233. Readily available.

Nuclear Waste Policy Act of 1982. 42 U.S.C. 10101 et seq. TIC: 241722.

CHAPTER 2. CONFORMANCE WITH TECHNICAL CRITERIA

This chapter contains guidance for the authors of Chapter 2 of the License Application (LA). Chapter 2 of the LA summarizes technical information, details of which are presented in other chapters, and presents certain information that supports technical bases and processes related to the potential monitored geologic repository (MGR). By following this technical guidance document (TGD), the LA authors will produce a chapter that will contribute to the ability of the U.S. Nuclear Regulatory Commission (NRC) to determine with reasonable assurance that the repository can be safely operated. Authors preparing the LA shall use this document for guidance and must read the Introduction and Appendix B before writing their respective sections.

Proposed 10 CFR 63.24(b) (64 FR 8640) recognizes that all information may not be available at the time of docketing the LA. The guidance in this chapter adopts a differentiated approach for providing the information required at construction authorization (CA) as opposed to that information required at license to receive and possess high-level radioactive waste (HLW) in the cases where information is expected to be different. Information in any section that is identified as "Information Required at the Time of Construction Authorization" will be provided at the time the LA is submitted for docketing. The information at the CA stage will be replaced with additional information developed since docketing the LA. The additional information is identified as, "Information Required at the Time of Update to the License Application to Receive and Possess."

2. PURPOSE AND SUMMARY

State that the purpose of Chapter 2 is to provide an overview of technical information related to the design of the potential MGR and that more detail is provided elsewhere in the LA. State that Chapter 2 also provides information regarding the location in the LA where the information identified in subsections of proposed 10 CFR 63.31(a) (64 FR 8640) is discussed in detail.

Briefly discuss the organization of Chapter 2, including the major topics covered (e.g., use and application of industry standards, classification analysis). State that Chapter 3 includes information regarding the natural features and processes of the site that are expected to act as natural barriers to radioactive transport as well as a description of the site geologic and climate evolution. Explain that detailed information regarding the design of individual MGR structures, systems, and components (SSCs) is contained in Chapters 4, 5, 6, and 9 of the LA. Also, explain that Chapters 5 and 6 contain the design description of the engineered barrier system (EBS). Explain that Chapter 7 contains information related to the preclosure radiological safety assessment, that Chapter 8 contains information related to the performance of the repository after permanent closure, and that Chapter 12 contains information related to the performance confirmation program. State that Chapter 10 contains a description of how the concept of "as low as is reasonably achievable" will be incorporated into the design of the repository.

Information Required at the Time of Construction Authorization–Note that proposed 10 CFR 63.31(a) (64 FR 8640) allows the NRC to authorize construction of a geologic repository operations area at the Yucca Mountain site if the NRC determines that there is reasonable assurance that the proposed types and amounts of radioactive materials described in the LA can be received, possessed, and disposed of at the potential MGR without unreasonable risk to the

health and safety of the public. State that the LA provides the information that supports such a determination by the NRC. In table or bullet form, list each of the subsections of proposed 10 CFR 63.31(a) (Subsections (1) through (5)) (64 FR 8640) and Interim Guidance Section 31(a)(6) (Dyer and Horton 1999), and indicate the chapters in the LA where that information is addressed.

State (if correct) that the U.S. Department of Energy (DOE) has resolved, at the staff level, all NRC questions on prelicensing submittals related to repository safety and performance with regard to use of, and conformance with, technical criteria. Identify any exceptions and discuss the status of resolution of these exceptions. Present an evaluation of the impact of unresolved questions, including comments or concerns identified during prelicensing interactions and reviews that might prevent the conduct of a meaningful review and a decision regarding CA within the three-year statutory time period. Concerns requiring a longer resolution time, such as new or additional testing or development of new or revised analytical methods, are examples of this type of objection. From the DOE perspective, discuss why unresolved safety questions, individually or in combination, should not prevent the NRC from conducting a meaningful compliance review or from making a decision regarding CA within the three-year statutory review period.

Draw conclusions with regard to the adequacy of the site information that was collected to support repository performance analyses. Support these conclusions, using direct quotations, by providing "preliminary comments" from the NRC regarding the sufficiency of the site characterization analysis that was conducted as addressed in the Nuclear Waste Policy Act of 1982 (NWPA) (see 42 U.S.C. 10134(a)(1)(E)).

Information Required at the Time of Update to the License Application to Receive and Possess–State, if applicable, that unresolved questions identified at the time of CA have been resolved, and identify any exceptions. If unresolved questions still remain, provide a basis for the issues to remain unresolved during receipt of HLW.

Present an evaluation of all unresolved questions, including comments or concerns identified during prelicensing interactions and reviews, and provide plans for resolution which will allow the conduct of a meaningful review and a decision regarding issuance of a license to receive and possess source, special nuclear, and by-product material at the potential MGR. Discuss (and cross-reference to Chapter 12) the role of the performance confirmation program in dealing with uncertainties remaining at the time of licensing.

2.1 CONFORMANCE WITH TECHNICAL CRITERIA IN PROPOSED 10 CFR 63 (64 FR 8640)

State that the purpose of this section is to provide an overview of how the engineered and natural systems comply with the technical criteria (performance objectives) contained in proposed 10 CFR 63.111(b)(1) (64 FR 8640) and Interim Guidance Section 111(b)(2) (Dyer and Norton 1999) through permanent closure and in proposed 10 CFR 63.113(a) and proposed 10 CFR 63.113(b) (64 FR 8640) after permanent closure. State that it also summarizes how the integrated safety analysis (ISA) and the performance assessment, specified in proposed 10 CFR 63, Subpart E (64 FR 8640), are addressed in the LA.

Proposed 10 CFR 63.31(a)(2) (64 FR 8640) states that the NRC may authorize construction of the potential MGR if it determines, among other things, that the site and design comply with the performance objectives and requirements contained in proposed 10 CFR 63 Subpart E (64 FR 8640). Discuss briefly and generally how the features of the engineered and natural systems result in compliance with the technical criteria (performance objectives) contained in proposed 10 CFR 63.111(b)(1) (64 FR 8640) and Interim Guidance Section 111(b)(2) (Dyer and Norton 1999). Discuss briefly and generally how the features of the engineered and natural systems result in compliance with the technical criteria (performance objectives) contained in proposed 10 CFR 63.113(a) and proposed 10 CFR 63.113(b) (64 FR 8640). Include a general description of the site and potential MGR design, emphasizing how aspects of the design integrate with the natural systems to meet these objectives. Refer to Section 2.2 to streamline this discussion and avoid repetition.

Proposed 10 CFR 63.21(c)(2) (64 FR 8640) requires that an ISA of the potential MGR be provided that covers the period before permanent closure to ensure compliance with proposed 10 CFR 63.111(a) (64 FR 8640). Introduce the concept of the ISA and its purpose. Make the point that individual elements of the ISA are discussed in various locations throughout the LA. State that the ISA has been performed and that it meets the requirements specified in proposed 10 CFR 63(c)(2) (64 FR 8640). State that the ISA demonstrates that the requirements of proposed 10 CFR 63.111(a) (64 FR 8640) will be met and that the potential MGR design meets the requirements of proposed 10 CFR 63.111(b)(1) (64 FR 8640) and Interim Guidance Section 111(b)(2) (Dyer and Horton 1999). Provide an overview discussion of how the ISA is addressed in the LA. Include information on where the various major elements of the ISA are addressed in the LA. Include in this discussion a statement that the demonstration of compliance with Interim Guidance Section 112(e) (Dyer and Horton 1999) is presented in Chapters 2, 4, 5, 6, 7, and 9 with the following allocations:

- The analysis performed to identify the SSCs important to safety is described in Section 2.4.
- The controls that are relied on to limit or prevent design basis events (DBEs) or mitigate their consequences are identified and described in individual sections of Chapters 4, 5, 6, 7, and 9.
- Special requirements or measures taken to ensure the availability of identified safety systems are identified in Chapters 4, 6, 7, and 9.

Provide a table (or other cross-referencing display) that cross-locates the following sections and related subsections to the chapters and sections in the LA where those regulations are addressed: proposed 10 CFR 63.111(a), proposed 10 CFR 63.111(b)(1), Interim Guidance Section 111(b)(2), proposed 10 CFR 63.111(c)(1), proposed 10 CFR 63.111(c)(2), proposed 10 CFR 63.112(a), Interim Guidance Section 112(b), proposed 10 CFR 63.112(c), proposed 10 CFR 63.112(d), and Interim Guidance Section 112(e) (Dyer and Horton 1999 and 64 FR 8640).

Interim Guidance Section 63.21(c)(7) (Dyer and Horton 1999) requires that the LA include an assessment of the performance of the potential MGR covering the period after permanent

closure, as required by proposed 10 CFR 63.113(c) (64 FR 8640). Introduce the total system performance assessment that was performed to meet this requirement. State that the results of the assessment required by proposed 10 CFR 63.113(c) (64 FR 8640) are described in Chapter 8 of the LA. State that the assessment described in Chapter 8 uses the reference biosphere and the critical group specified in proposed 10 CFR 63.115 (64 FR 8640) and demonstrates the ability of the potential MGR to limit exposures to those specified in proposed 10 CFR 63.113(b) (64 FR 8640).

Proposed 10 CFR 63.21(c)(8) (64 FR 8640) requires that the LA include an assessment of the ability of the potential MGR to limit radiological exposures in the event of a limited human intrusion into the EBS, as required by Interim Guidance Section 113(d) (Dyer and Horton 1999). State that the total system performance assessment includes an assessment of the ability of the potential MGR to continue to isolate waste over the long term in the event of limited human intrusion, as required by Interim Guidance Section 113(d) (Dyer and Horton 1999). State that the results of this assessment are described in Chapter 8 of the LA and that the assessment meets the requirements of proposed 10 CFR 63.114 (64 FR 8640). State that this assessment uses the reference biosphere and the critical group specified in proposed 10 CFR 63.115 (64 FR 8640) and demonstrates the ability of the potential MGR to continue to perform acceptably following human intrusion, as addressed in Interim Guidance Section 113(d) (Dyer and Horton 1999).

2.2 WASTE CONTAINMENT AND ISOLATION

Introduce the section by stating that it summarizes the comprehensive postclosure safety case for the potential MGR. Discuss how containment and isolation of waste at Yucca Mountain will be accomplished by a natural and engineered multiple barrier system (proposed 10 CFR 63.113(a) [64 FR 8640]) to meet the postclosure performance objective of proposed 10 CFR 63.113 (64 FR 8640). At a summary level, discuss how the repository relies on natural and engineered barriers to meet performance standards, and provide a reference to other sections of the LA for more information.

Discuss each element of the postclosure safety case: performance assessment, design margin and defense in depth, potentially disruptive processes and events, natural analogues, and a performance confirmation plan. Include tables and figures as appropriate to present the discussions and arguments.

2.3 USE OF U.S. NUCLEAR REGULATORY COMMISSION TECHNICAL POSITIONS, U.S. NUCLEAR REGULATORY COMMISSION REGULATORY GUIDES, AND INDUSTRY STANDARDS

Proposed 10 CFR 63.21(c)(3) (64 FR 8640) requires that the safety analysis report (Chapters 2 through 13 of the LA) include information relative to codes and standards that the DOE proposes to apply to the design and construction of the potential MGR. State that the purpose of this section is to summarize technical positions, regulatory guides, and industry standards that the DOE intends to use at the repository and the extent to which they have been or will be used for design and construction of each system. Provide a brief description of the process that the DOE used to assess the applicability of technical positions, regulatory guides, and industry codes and standards to potential MGR activities.

In the text, refer to the tables that are required in the following paragraphs, providing a brief description and purpose of each table. The information for the tables will come from other chapters of the LA to create a composite summary.

Provide a table identifying the NRC technical positions on the repository that the DOE intends to use (this information can be obtained from Chapters 4, 5, 6, and 9 in discussions of individual systems and processes). In the table, identify the titles, numbers, and revision numbers of those positions. For each technical position listed, identify in the table those sections of the LA to which the position applies and the extent to which the technical position is implemented (e.g., entire document, selected sections).

Describe the DOE philosophy of implementing regulatory guides at the potential MGR. State that, in some cases, the DOE will comply completely with positions in regulatory guides; however, in other cases, the DOE will comply with selected positions, depending on evaluation of applicability of the regulatory guides.

Present a table that lists NRC regulatory guides important to establishing the safety case for the potential MGR and other important regulatory guides that may contribute to NRC review of the LA. These guides are identified in other chapters of the LA. For each, give the title, number, and revision number and state the extent to which the repository program complies with that regulatory guide (e.g., partially compliant or fully compliant). Include in the table a cross-reference to sections in the LA where the use of the regulatory guide is discussed. Develop this table based on information contained in individual LA chapters that describe how each applicable regulatory guide is applied.

Discuss how industry codes and standards are to be used for the design and construction of SSCs. Describe how these codes and standards define acceptable methods and parameters for designing the mechanical, electrical, civil, and nuclear systems provided for the repository. Provide a reference to individual sections of the design chapters (Chapters 4, 5, 6, and 9) for information about principal codes and standards implemented for SSCs.

Identify the Code of Record (governing code and edition) to which the design and construction of the repository and waste package are committed.

2.4 CLASSIFICATION AND GRADING OF STRUCTURES, SYSTEMS, AND COMPONENTS

State that the purpose of this section is to describe the classification and grading of SSCs at the potential MGR.

Interim Guidance Section 112(e) (Dyer and Horton 1999) requires, in part, that the ISA of the potential MGR include an analysis of the performance of the major surface and subsurface design SSCs to identify those that are important to safety. Introduce this section by stating that this requirement has been met by performing a classification analysis.

Explain the purpose of classifying SSCs.

State that the *Quality Assurance Requirements and Description* (QARD) (DOE 1998, Section 2.2.2) document governs the classification process. Discuss the process for identifying those SSCs important to safety. Identify industry guidance documents used, and the extent to which each was followed, in performing the classification analysis. Clearly define each classification. Explain that the Quality Level (QL) 1 classification includes permanent items important to radiological safety and important to waste isolation. Use tables and figures, as appropriate, to explain the defining differences among the classifications.

Provide a table that lists each MGR systems and subsystems and the related quality classification. State that discussions in Chapters 4, 5, 6, and 9 provide additional information on the functions, including controls, of SSCs important to safety that are relied upon to limit, prevent, or mitigate the consequences of DBEs. Measures taken to ensure the availability of identified safety systems are also discussed in these chapters.

State that the potential MGR uses a graded approach to implementation of the quality assurance (QA) program related to SSCs, as addressed in and allowed by the *Quality Assurance Requirements and Description* (DOE 1998, Section 2.2.4). Identify the industry guidance documents used, and the extent to which each was followed, in defining and implementing the graded QA approach. Briefly summarize the graded QA approach used and its relationship to the defined quality levels. Refer to Chapter 1 for a description of the QA program, and identify the elements of the QA program that apply to each classification category. Make the point that all elements of the QA program apply to QL 1 items, which include those important to radiological safety and to waste isolation. Provide tables or figures, if needed, to describe the graded program.

2.5 STRUCTURES, SYSTEMS, AND COMPONENTS IMPORTANT TO SAFETY

State that the purpose of this section is to provide an overview description of how the major SSCs of the potential MGR perform their operational functions and achieve their design functions.

Using referrals to Section 2.4, relate the phrase "important to safety" to QLs. Provide an overview of the operational activities of receiving, packaging, and emplacing waste. Describe the functions of the major repository SSCs only to the extent necessary to support the overview. This overview is intended to be brief and may incorporate selected general information in the *Reference Design Description for a Geologic Repository* (CRWMS M&O 1999a).

Discuss the approach taken in the LA (Chapters 4, 5, 6, and 9) regarding the level of design detail provided for the various engineered SSCs based on the potential impact to the health and safety of the public and workers (i.e., the quality classification). Justify the position taken regarding the adequacy of the differing amount of detail. Base the discussion on the *Level of Design Detail Necessary for the License Application for Construction Authorization* (CRWMS M&O 1999b).

State that Chapter 7 of the LA contains a discussion of the evaluation of Categories 1 and 2 DBEs. Provide a reference to Section 7.2 for a description of the approach used for assessing the radiological safety of the potential repository until permanent closure and to Section 7.5 for

information regarding screening, selection, and systematic review of events. Relate the identification of DBEs to the classification of SSCs discussed in Section 2.4 and to SSC design.

Briefly summarize the design and design bases of major Quality Levels 1 and 2 SSCs. Briefly discuss how these major SSCs accomplish their design functions. Refer to Chapters 5 and 6 for a discussion of the design bases of the EBS and their relation to the principal design criteria (proposed 10 CFR 63.21(c)(4)(ii) [64 FR 8640]). Provide a reference to the design chapters of the LA (Chapters 4, 5, 6, and 9) for information about design bases, design criteria, and specific codes and standards applicable to design and construction of each system (proposed 10 CFR 63.21(c)(3) [64 FR 8640]). Provide a reference to Section 2.3 for information about codes and standards applied to the design of potential MGR systems.

2.6 REFERENCES

The following references were used to develop this chapter of the TGD. For the LA, this section will contain the references used to develop this chapter of the LA.

References Cited

CRWMS (Civilian Radioactive Waste Management System) M&O (Management and Operating Contractor) 1999a. *Reference Design Description for a Geologic Repository.* B00000000-01717-5707-00002 REV 02. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990301.0225.

CRWMS M&O 1999b. *Level of Design Detail Necessary for the License Application for Construction Authorization.* B00000000-01717-1710-00003 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990708.0065.

DOE (U.S. Department of Energy) 1998. *Quality Assurance Requirements and Description.* DOE/RW-0333P, Rev. 8. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.19980601.0022.

Dyer, J.R. and Horton, D.G. 1999. "Interim Guidance Pending Issuance of New Nuclear Regulatory Commission (NRC) Regulations for Yucca Mountain, Nevada." Letter from J.R. Dyer (DOE/YMSCO) to D.R. Wilkins (CRWMS M&O), June 18, 1999, OL&RC:AVG:1435, with enclosure, "Interim Guidance Pending Issuance of New NRC Regulations for Yucca Mountain." ACC: MOL.19990712.0039.

Codes, Standards, and Regulations

64 FR (Federal Register) 8640. Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, Nevada. Proposed rule 10 CFR 63. Readily available.

Nuclear Waste Policy Act of 1982. 42 U.S.C. 10101 et seq. TIC: 241722.

INTENTIONALLY LEFT BLANK

CHAPTER 3. SITE CHARACTERISTICS

State the objective of this chapter of the License Application (LA), which is to provide a description of the characteristics of the Yucca Mountain site and the surrounding region. The authors will produce a chapter that will demonstrate that the natural systems of the site have been characterized adequately to support repository design and performance assessment (PA). This chapter of the technical guidance document (TGD) provides guidance for the authors of Chapter 3 of the LA to describe the natural systems that comprise the natural features, events, and processes (FEPs) of the site. By following the TGD, the LA authors will produce a chapter that will enable the U.S. Nuclear Regulatory Commission (NRC) to determine, with reasonable assurance, that the natural FEPs of the site will perform well enough with the engineered barrier system to ensure that the repository can be safely operated and will perform as expected after permanent closure. Authors preparing the LA shall use this document for guidance and must read the Introduction and Appendix B before writing their respective sections.

The guidance presented in this chapter applies for docketing of the LA at the time of construction authorization and at the time of updating the LA to receive and possess high-level radioactive waste. All the information delineated in this chapter must be presented in the LA at the time of docketing the LA. The information will be confirmed to be correct or brought up to date as appropriate when the LA update is submitted.

Compliance program guidance packages have been prepared related to the contents of this chapter. Ensure all applicable requirements, as identified in the appropriate compliance program guidance packages are addressed.

3. PURPOSE AND SUMMARY

State the purpose of Chapter 3 of the LA, which is to provide a sufficiently detailed description of the natural FEPs of the site to document that they have been adequately characterized to support design and PA needs. State that the information in this chapter is separated into major natural systems to facilitate demonstration of compliance with the regulatory requirements for a description of the site as outlined in proposed 10 CFR 63.21(c)(1)(i) (64 FR 8640).

Describe the organization of Chapter 3 of the LA.

Statutory and Regulatory Bases for Site Characterization–State the purpose of this section of Chapter 3, which is to summarize the statutory and regulatory bases for site characterization (i.e., the Nuclear Waste Policy Act of 1982, as amended; 10 CFR 60; proposed 10 CFR 63 [64 FR 8640]). (Refer to descriptions of the statutory and regulatory bases for site characterization in the *Site Characterization Plan* [DOE 1988].)

Overview of Site Characterization–State the purpose of this section of Chapter 3, which is to summarize the scope of the investigations and the work done to characterize geography, demography, geology, hydrology, geochemistry, climatology, and meteorology. Include a summary of work done to develop the three-dimensional (3-D) Integrated Site Model and the supporting Geologic Framework Model, the Rock Properties Model, and the Mineralogic Model. Also summarize the work done to develop the Unsaturated Zone (UZ) Flow and Transport

Model and the Saturated Zone (SZ) Flow and Transport Model, and to characterize seismic and volcanic hazards. Summarize how the information about site characteristics supports design and PA. Refer to discussions of adequacy of individual testing programs in other sections of this chapter.

Key Natural Features–State the purpose of this section of Chapter 3, which is to describe the characteristics of the site and identify features and processes that are expected to act as natural barriers (proposed 10 CFR 63.113(a) and proposed 10 CFR 63.114(h) [64 FR 8640]) or that could materially affect the design or performance of the repository system. Reference more detailed discussions of potential hydrologic barriers provided in the description of the hydrology system and the potential geochemical barriers discussed with the geochemical system. Summarize how information about these key natural features has been integrated into the design of the repository system and into the total system performance assessment (TSPA). Provide cross-references to discussions of natural FEPs that are expected to affect design of the surface facilities (Chapter 4), waste package (Chapter 5), or repository (Chapter 6). Identify natural features and processes that are expected to act as barriers (proposed 10 CFR 63.114(i) and proposed 10 CFR 63.114(j) [64 FR 8640]), and provide cross-references to discussions in Chapter 8 of features and processes that are expected to act as natural barriers. Briefly summarize the safety significance of these key natural features and processes in terms of how these features and processes affect the safety of the repository.

3.1 GEOGRAPHY AND DEMOGRAPHY

State the purpose of this section of Chapter 3, which is to summarize the geography and demography of the region around Yucca Mountain. Identify the topics to be discussed in this section and briefly discuss the organization of the section. Describe the geography and demography study program. Discuss why the program was adequate to support characterization of the geography and demography, and why it was adequate to provide information needed to describe some reference biosphere and critical group parameters for PA.

Ensure all applicable requirements, as identified in the appropriate compliance program guidance package, are addressed.

3.1.1 Data Sources and Quality

State the purpose of this section of Chapter 3, which is to describe the sources of data and the quality assurance (QA) status of data that support the discussions of geography and demography. State that geographic and demographic data collected by the Yucca Mountain Site Characterization Project (YMP) and used as the basis for the repository safety case were collected and documented, or qualified in accordance with the QA program described in Section 1.5 of this LA. Discuss the variability and uncertainty of data and information, as well as the propagation of errors. Discuss the extent to which data were collected and documented and analyses were developed and documented under acceptable QA procedures. Discuss the variability and uncertainty of data and information, as well as the propagation of errors. Discuss the use of data from non-YMP sources. If non-YMP data are considered "accepted data" as defined in the *Quality Assurance Requirements and Description* (QARD) (DOE 1998, Glossary), describe the basis for considering the data "accepted."

Describe the representativeness of data provided by the geography and demography program, and explain that Chapter 8 will describe the measures used to support the models and abstractions in compliance with proposed 10 CFR 63.114(g) (64 FR 8640).

3.1.2 Site Geography

State the purpose of this section of Chapter 3, which is to describe the location of the geologic repository operations area (GROA) with respect to the boundaries of the site. Describe the size, shape, dimensions, and boundaries of the GROA. Provide maps, at appropriate scales, showing the location of the site, the geometric properties of the site, and the geometric properties of the GROA. Show the location of the GROA relative to the site boundaries. Provide geographic locations of key GROA facilities using longitude and latitude, State Plane, and Universal Transverse Mercator coordinate systems. As appropriate, provide cross-references to Chapter 4 for a more detailed description of the GROA. Provide cross-references to discussions of land ownership and control in Chapter 13.

State that the information in this section demonstrates compliance with the regulatory requirement to provide information about the location of the GROA with respect to the boundaries of the site (proposed 10 CFR 63.21(c)(1)(i) [64 FR 8640]).

3.1.3 Site Demography

State the purpose of this section of Chapter 3, which is to discuss current population statistics (distribution and density), lifestyles, and socioeconomic characteristics of the site and the surrounding region. As appropriate and needed, provide historical data and discuss historic population trends. Provide figures showing the population distribution and density. Discuss how demographic and lifestyle information was collected, analyzed, and reported.

Provide cross-references to discussions of similarities between the demographic and lifestyle characteristics of the critical group and those of the current demographic and lifestyle characteristics of the population in the region surrounding Yucca Mountain (proposed 10 CFR 63.21(c)(1)(iv) and proposed 10 CFR 63.115(b)(2) [64 FR 8640]) and the characteristics of the reference biosphere in Chapter 8. Provide cross-references to water use information given in the hydrology section (Section 3.5.5.2), which is needed to support the descriptions of the critical group and the reference biosphere.

3.2 REGIONAL AND SITE GEOLOGIC SETTING

State the purpose of this section of Chapter 3, which is to summarize the regional and site geological setting of Yucca Mountain. Identify the topics to be discussed in this section and briefly discuss the organization of the section. Describe the geology testing program, and discuss why it was adequate to support the characterization of the geologic system and why it was adequate to provide information needed for design and PA. Ensure all applicable requirements, as identified in the appropriate compliance program guidance package, are addressed.

Explain that Chapter 8 will describe the measures used to support the models and abstractions in compliance with proposed 10 CFR 63.114(g) (64 FR 8640).

3.2.1 Overview

State the purpose of this section of Chapter 3, which is to summarize the main elements of the regional and site geologic system and describe how the regional geology provides a framework for the description of the site geology. Summarize alternate interpretations of geologic FEPs. Compare them in terms of their impact on the conceptual framework for the site, and discuss why the U.S. Department of Energy (DOE) does not subscribe to the alternative interpretation(s). State that the information in this section complies with the requirement that the Safety Analysis Report (SAR) contain "information regarding the geology...of the site...[and] geomechanical properties and conditions of the host rock" (proposed 10 CFR 63.21(c)(1)(ii) [64 FR 8640]).

3.2.2 Data Sources and Quality

State the purpose of this section of Chapter 3, which is to describe the sources of data and the QA status of data that support the discussions of regional and site geology. State that site geologic data used as the basis for the repository safety case were collected and documented, or qualified in accordance with the QA program described in Section 1.5 of this LA and associated implementing procedures [RDTME 3.1.1] [RDTME 3.3.1] (NRC 1998a), [SD&S 1.1.1] [SD&S 1.2.1] [SD&S 2.1] [SD&S 3.1.1] [SD&S 4.1.1] (NRC 1998b), [IA 1.9] (NRC 1998c). Discuss the variability and uncertainty of data and information, as well as the propagation of errors. Discuss the use of data from non-YMP sources. If non-YMP data are considered "accepted data" as defined in the QARD (DOE 1998, Glossary), describe the basis for considering the data "accepted." Also, and as appropriate, summarize the use of expert elicitation (proposed 10 CFR 63.21(c)(10) [64 FR 8640]), and discuss the extent to which expert elicitations were conducted in accordance with the guidance in NUREG-1563 (NRC 1996) or other accepted approaches [SD&S 1.1.2] [SD&S 1.2.2] [SD&S 2.2] [SD&S 3.1.2] [SD&S 4.1.2] [IA 1.8] (NRC 1998c). In these discussions, emphasize that expert elicitations were conducted in conformance with the QARD (DOE 1998).

Describe the representativeness of data provided by the regional and site geology program, and explain that Chapter 8 will describe the measures used to support the models and abstractions in compliance with proposed 10 CFR 63.114(g) (64 FR 8640).

3.2.3 Regional Geologic Setting

State the purpose of this section of Chapter 3, which is to describe the regional geologic setting of the Yucca Mountain site. Describe the regional stratigraphy, structure, and general physiographic features and processes of the geologic setting that form the geologic framework for the site. Summarize the evolution of the geologic setting in terms of the present knowledge of natural processes (Interim Guidance Section 114(l) [Dyer and Horton 1999]).

3.2.4 Regional Tectonic Models

State the purpose of this section of Chapter 3, which is to describe the viable tectonic models applicable to the region surrounding the Yucca Mountain site. Describe the viable tectonic models of the Yucca Mountain region and summarize the bases for their selection. Discuss the range of published tectonic models and supporting geological, geophysical, and modeling data and the results that were considered in identifying the viable tectonic models for Yucca Mountain and the surrounding region. Describe the technical bases of the viable models with regard to existing geophysical, geological, seismological, and geodetic data. Describe the crustal conditions, structural style, and deformation mode at the site and within the tectonic framework of the southern Great Basin. Discuss how model elements and associated uncertainties are incorporated in each viable tectonic model. Discuss the implications of viable tectonic models in the evaluation of igneous activity and seismicity, and discuss how regional tectonic models are used in evaluating seismic and volcanic hazards. Also, discuss how tectonic models are used in the consequence analyses presented in Chapter 8.

Provide a summary-level demonstration that the information in this section addresses aspects of tectonic models of the Yucca Mountain region [SD&S 4] (NRC 1998b):

- Discuss how a set of viable tectonic models for Yucca Mountain and the surrounding region has been developed from the numerous published tectonic models and supporting geological, geophysical, and modeling data and results. Provide a reasonable explanation of the bases for selection of viable tectonic models, purposes of each model, and demonstrations that each model is internally consistent with the appropriate structural style and deformation mode and compatible with the tectonic framework of the southern Cordillera and the Basin and Range province [SD&S 4.1.3] (NRC 1998b).

- Discuss how viable tectonic models are consistent with existing geophysical, geological, seismological, and geodetic data, and include reasonable explanations of how data that were inconsistent with the model were accounted for. For example, describe how the appropriate data have been evaluated, including, but not restricted to: geophysical (gravity, magnetics, paleomagnetics, seismic refraction/reflection, teleseismic), geological (structural, geothermal, geochronological), seismological (historic seismicity, crustal condition, paleoseismicity), and geodetic (global positioning system, trilateration survey, level line survey). Discuss the treatment of data inconsistencies [SD&S 4.1.4] (NRC 1998b).

- Show that viable tectonic models depict the tectonic, structural, and seismic elements and the uncertainties associated with the quantification of each element critical for the intended purpose of each model. For example, scaling tools, including but not restricted to geologic maps, block diagrams, and restorable cross sections, have been adequately used, and reasonable interpretations of geologic, geometric, kinematic, and mechanical relationships used to constrain the key uncertainties have been adequately evaluated [SD&S 4.1.5] (NRC 1998b).

3.2.5 Site Geology

State the purpose of this section of Chapter 3, which is to summarize the principal features and processes of site geology. Summarize the principal features and processes of the site physiography, stratigraphy, and structure. Identify the geologic FEPs that are expected to materially affect design or compliance with the performance objectives for the repository (proposed 10 CFR 63.21(c)(5) [64 FR 8640]). Also identify those geologic features and processes that are expected to act as barriers (proposed 10 CFR 63.114(i) and proposed 10 CFR 63.114(j) [64 FR 8640]), and provide cross-references to discussions of barriers in Chapter 8. State the assumptions and describe the data that provide the bases for conclusions about site geology. Also, discuss alternative interpretations and views, and explain why the DOE does not subscribe to the alternatives. Briefly summarize how characteristics of the principal geologic features and processes of the site are expected to affect the evaluation of the safety of the site.

3.2.6 Surficial Geology and Quaternary Stratigraphy

State the purpose of this section of Chapter 3, which is to describe the site surficial geology and Quaternary stratigraphy. Describe the Quaternary stratigraphy and surficial features and processes found at and near Yucca Mountain. Summarize the paleoenvironmental history of Yucca Mountain, describe the model of landscape response to climate change at Yucca Mountain, and, as appropriate, provide references to descriptions in *Evaluation of the Potentially Adverse Condition "Evidence of Extreme Erosion During the Quaternary Period" at Yucca Mountain, Nevada* (YMP 1993). Also, as appropriate, summarize additional information about the landscape response that has been developed since the report was completed (Bell 1996). Summarize alternate interpretations of the near-surface calcite-silica deposits and the technical bases for DOE conclusions regarding these deposits.

3.2.7 Site Stratigraphy

State the purpose of this section of Chapter 3, which is to describe the pre-Quaternary stratigraphy of the site. Describe the Tertiary and Paleozoic stratigraphy and lithologic framework for the site. Describe the stratigraphic or lithologic units within and surrounding the repository host horizon and discuss their origins. Describe the key features used to discriminate stratigraphic or lithologic units, such as mineralogic and petrographic features, including texture, fabric, grain size, percent and size of cavities, degree of welding, glass content, and degree of devitrification. Describe the vertical and lateral variations in compositional or physical characteristics. Cross-reference discussions of mineralogic and rock properties in the 3-D Integrated Site Model section (see Section 3.2.10) of geomechanical units in the geoengineering properties section (see Section 3.2.9) and of hydrogeologic properties in the hydrology sections (see Sections 3.5.5.1 and 3.5.5.2).

3.2.8 Site Structural Geology

State the purpose of this section of Chapter 3, which is to describe the structural geology of the site. Describe the structural features of the site. Discuss the results of surface, subsurface, and borehole geological and geophysical investigations pertaining to site structure. Discuss the

sensitivities and resolutions of the various techniques as they apply to identifying structures or determining rates of deformation. Discuss the geologic structure of the site area and the central block. Reference additional discussions in the Integrated Site Model section (see Section 3.2.10) of how site structure information is incorporated in the model. Briefly summarize structural controls on fracturing, and cross-reference discussions of fractures and fracture properties in the site geoengineering properties section (see Section 3.2.9).

3.2.9 Site Geoengineering Properties

State the purpose of this section of Chapter 3, which is to identify the geologic media that will be used to construct the repository and summarize the geoengineering properties of the site that are expected to affect repository design, construction, operation, or postclosure performance. Describe the geologic media that will be used to construct the repository (proposed 10 CFR 63.21(c)(3) [64 FR 8640]). Briefly summarize how the site geoengineering properties are expected to affect the evaluation of the safety of the site. Provide cross-references to descriptions of stratigraphic units in Section 3.2.7. State that this section of Chapter 3 discusses the pattern of fractures in the rock and other discontinuities of the rock mass. State also that Chapter 8 will include an assessment of the anticipated response of the geomechanical, hydrogeologic, and geochemical systems to the range of design thermal loadings under consideration, given the pattern of fractures, and other discontinuities and the heat transfer properties of the rock mass and groundwater (proposed 10 CFR 63.21(c)(6) [64 FR 8640]).

Summarize the geoengineering properties determined for the site and summarize the mechanical and thermal properties for each unit. Discuss how the information in this section addresses aspects of thermal-mechanical effects on design of the underground facility by providing site-specific thermal and mechanical properties, and the spatial distribution of those properties [RDTME 3.1.3] (NRC 1998a). Summarize evaluations of the effects of anisotropy, lithophysae, porosity, density, and water saturation on the geoengineering properties of rock and alluvial units. Cross-reference discussions of the spatial distribution of properties in the Integrated Site Model (Section 3.2.10). Discuss the testing and sampling procedures used to determine geomechanical and thermal properties. Discuss the surface, subsurface, and borehole studies of fractures.

Describe fracture intensity, connectivity, spatial variability, and structural controls on fracture network properties. Describe the design and TSPA uses of information about distributions, spacings, densities, and orientations of fractures and discontinuities, including lithophysae. Discuss how this information addresses thermal-mechanical effects on design of the underground facility by providing information about potential effects of lithophysae on rock strength [RDTME 3.1.14] (NRC 1998a). Discuss the significance of structural and lithologic controls on fracture network properties. Discuss the significance of fracture aperture infillings as they affect geomechanical properties. Discuss excavation characteristics of the rock mass, including implications for the stability of openings over time. Include consideration of the effects of the distribution of joint patterns, spacings, and orientations as these parameters affect the intensity and distribution of ground movement (rock deformations, collapse, and other changes that may affect the integrity or geometrical configuration of openings within the underground facility.

Summarize how information about laboratory and rock mass properties of the thermal-mechanical stratigraphic units of the site is used.

If it has not already been provided, explain how the information in this section addresses the need to identify the viable models of fractures and structural controls on flow at Yucca Mountain. Include discussions of aspects of fractures, including the following [SD&S 3] (NRC 1998b):

- Show that distribution and geometric characteristics (e.g., orientations, spacing, clustering, abutting relationships, interconnectedness, apertures, lengths) of fractures were adequately determined. For example, provide a comprehensive unit-by-unit description of fractures that captures lateral and vertical variability of fracture development and interconnectivity throughout the Tertiary volcanic rock sequence and pre-Tertiary rock sequence at Yucca Mountain. Summarize the expected impacts of fractures and faults on hydrology-related aspects of repository performance, and provide cross-references to discussions in Chapter 8 [SD&S 3.3] (NRC 1998b).

- Demonstrate that the mechanisms for fracture generation, including development of cooling joints, tectonic joints, and unloading joints were adequately explained in a way that is consistent with the evolution of the applicable regional or local stress field and detailed enough to assess aspects of fractures and faults that affect repository performance [SD&S 3.4] (NRC 1998b).

- Show that subsequent modifications of fractures by dissolution, precipitation, wall rock deformation, and other fracture-filling processes (e.g., deposition of water-entrained particles) were adequately constrained. For example, discuss characteristics of fracture-filling materials that would affect fracture absorption of surface water, role of fractures as barriers to flow, isolation of fracture water from host rock because of armoring of wall rock surfaces by fracture coatings, and dissolution along fractures that may enhance hydraulic conductivity. Summarize the expected effects of these changes in fracture characteristics on repository performance [SD&S 3.5] (NRC 1998b). Provide cross-references to discussions of hydrologic properties in Section 3.5.5.1. Provide cross-references to Chapter 8 discussions of the role of fractures and changes in fracture properties on UZ flow and on repository performance.

- Indicate, using cross-references, that Chapter 8 contains discussions of potential current and future tectonically and thermally controlled alteration of fracture characteristics during the repository performance period, and that alterations of fracture characteristics have been adequately defined and accounted for in process-level models. For example, summarize and provide cross-references to evaluations in Chapter 8 of structural and tectonic models for contemporary or future changes to fracture characteristics (e.g., increases and decreases in fracture apertures) caused by in situ stress, contemporary strain accumulation, seismic and aseismic deformation events, or differential thermal expansion and contraction [SD&S 3.6] (NRC 1998b). Summarize how these parameters have been documented and propagated through flow and transport and total system models to assess aspects of fractures and faults that affect repository performance. Provide cross-references to discussions in Chapter 8.

State that the information in this section demonstrates compliance with the requirement that the SAR contain "information ...including [the] geomechanical properties and conditions of the host rock" (proposed 10 CFR 63.21(c)(1)(ii) [64 FR 8640]).

3.2.10 Three-Dimensional Integrated Site Model

State the purpose of this section of Chapter 3, which is to describe the 3-D Integrated Site Model. Describe the Integrated Site Model in terms of the 3-D Geologic Framework Model, the rock properties and mineralogic models, and the stratigraphic notebooks that provide input to the Integrated Site Model. Discuss the development of the Integrated Site Model and its component models in terms of data sources, assumptions, and approaches. Identify any baseline models or standards required for derivative process models.

Describe how the model captures and portrays site structural elements, variations in mineralogical and rock properties (e.g., distribution of zeolites, clays, silica polymorphs, and glass), and stratigraphic unit boundaries. Discuss how representative these data are. Discuss how the Integrated Site Model demonstrates that adequate understanding of these site features exists to support design and TSPA evaluations. To the extent applicable and appropriate, differentiate between assumptions, data, and alternate interpretations captured in the model.

3.2.11 Volcanism and Volcanic Hazards

State the purpose of this section of Chapter 3, which is to describe the volcanism and volcanic hazards of the site. Summarize the volcanic history of the site. Describe the basaltic igneous activity at the site and in the surrounding region, with emphasis on the Quaternary Period. Describe the investigations that were done to evaluate the volcanic hazard and provide references to the *Probabilistic Volcanic Hazards Analysis for Yucca Mountain, Nevada* (CRWMS M&O 1996) and other analyses and modeling or process-model reports, as appropriate. Summarize the volcanic hazard results and provide references to the probabilistic volcanic hazard analysis (CRWMS M&O 1996). Discuss how differing opinions of the NRC on the volcanic hazard have been considered, and discuss how conflicting information has been incorporated into the DOE hazard estimate. Discuss the potential for coupled volcanic and seismic events, as discussed in the *Probabilistic Seismic Hazards Analyses for Fault Displacement and Vibratory Ground Motion at Yucca Mountain, Nevada* (CRWMS M&O 1998). Identify the characteristics of the geologic setting, regional geologic structures, and tectonic activity that may control the locations of future volcanic activity or coupled volcanic-seismic activity. Summarize how the results of the probabilistic seismic hazard analysis (CRWMS M&O 1998) are included in the consequence analyses and in the TSPA, and provide references to relevant discussions in Chapter 8.

If not already discussed, provide a summary-level demonstration that the information in this section shows the following [IA 1] (NRC 1998c):

- Estimates of probability of igneous activity are based on past patterns of igneous activity in the Yucca Mountain region [IA 1.1] (NRC 1998c).

- Models of igneous activity are consistent with observed patterns of volcanic vents and related igneous features in the Yucca Mountain region [IA 1.3] (NRC 1998c).

- Parameters used in probabilistic volcanic hazard analyses, which are related to the recurrence rate of igneous activity in the Yucca Mountain region, the spatial variation in frequency of igneous events, and the area affected by igneous events, are technically justified and documented by the DOE [IA 1.4] (NRC 1998c).

- Models of igneous activity are consistent with tectonic models proposed by the NRC and the DOE for the Yucca Mountain region [IA 1.5] (NRC 1998c).

- The probability values used by the DOE in PAs reflect the uncertainty in DOE probabilistic volcanic hazard analyses [IA 1.6] (NRC 1998c).

- The values used (single values, distributions, or bounds on probabilities) are technically justified and account for uncertainties in probability estimates [IA 1.7] (NRC 1998c).

Explain that the consequences of igneous activity are addressed in Chapter 8, and provide a cross-reference to the Chapter 8 discussion [IA 2] (NRC 1998c).

3.2.12 Seismicity and Seismic Hazards

State the purpose of this section of Chapter 3, which is to describe the seismicity and seismic hazards at the site. Summarize the fault history and contemporary seismicity of the site. Summarize the seismic hazard methodology described in Topical Report YMP/TR-002-NP (YMP 1997). Describe the results of the probabilistic seismic hazard analysis (CRWMS M&O 1998), and provide cross-references to TSPA consequence analyses in Chapter 8. Include a brief discussion of how seismic hazard inputs are used to estimate rockfall potential [RDTME 3.2.3] (NRC 1998a) and the size distribution of rocks that may potentially fall [RDTME 3.2.4] (NRC 1998a). Summarize the use of expert elicitation to estimate the vibratory ground motion and fault displacement hazards and to evaluate uncertainty in the hazard estimate (proposed 10 CFR 63.21(c)(10) [64 FR 8640]). Discuss the conformance of the expert elicitation with the QARD (DOE 1998) and with guidance in NUREG-1563 (NRC 1996). Discuss how conflicting information has been incorporated into the hazard estimate. Identify the characteristics of the geologic setting, regional geologic structures, and tectonic activity that may control the locations of future seismic activity or coupled volcanic-seismic activity.

Explain that the information in this section discusses site-specific seismic hazards and seismic design bases, and thereby partially addresses requirements to have and use site-specific data in the repository design. Discuss the adequacy and availability of site-specific data to support design by indicating that adequate data are available now, are being gathered now, or there are plans for gathering such data during site characterization and before submittal of the LA [RDTME 2.6] (NRC 1998a). Show that hazard calculations have been adequately documented, are coherent, and are technically defensible.

If it has not already been done, provide a summary-level discussion of how the information in this section addresses Type I faults, fault displacement, and seismic hazards [SD&S 1-2] (NRC 1998b).

- Show that the faulting component within the vicinity of Yucca Mountain was adequately determined. For example, show that the DOE investigated all known faults within an adequate distance (100 km) from the site to ensure that all candidate Type I faults have been investigated [SD&S 1.1.3] (NRC 1998b).

- Show that the maximum earthquake for each candidate Type I fault was adequately determined. For example, show that the DOE used an appropriate and adequate fault length vs. magnitude relationship that tended not to underestimate the seismic hazard [SD&S 1.1.4] (NRC 1998b).

- Show that the maximum trace length of each candidate Type I fault was measured from acceptable sources. For example, show that the DOE relied on appropriate primary map sources and adequate interpretations of segmented faults that tended not to underestimate the seismic hazard [SD&S 1.1.5] (NRC 1998b).

- Show that peak ground motion acceleration for each Type I fault was adequately determined. For example, show that the DOE used appropriate and adequate attenuation models that tended not to underestimate the seismic hazard [SD&S 1.1.6] (NRC 1998b).

- Show that the shortest distance to site boundary of each Type I fault was adequately measured. For example, show that the DOE used appropriate (the latest version of the smallest scale) primary geologic map and site-boundary sources [SD&S 1.1.7] (NRC 1998b).

- Show that the geologic age of last movement of each Type I fault was adequately determined. For example, show that the DOE used appropriate and adequately conservative interpretations of evidence of fault movement during the Quaternary Period [SD&S 1.1.8] (NRC 1998b).

- Show that potential for future slip was adequately determined. For example, show that when low potential for future slip on a Type I fault was determined, the DOE used appropriate magnitudes and orientations of principal stresses and fault-orientations and adequately conservative interpretation of slip-tendency [SD&S 1.1.9] (NRC 1998b).

- Show that minimum trace length for a Type I fault to be considered in a fault displacement hazard analysis was adequately determined. For example, show that the DOE used appropriate historic seismic records and surface-rupture data to determine the minimum surface-faulting earthquake and to back-calculate the associated trace length [SD&S 1.1.10] (NRC 1998b).

- Show that the nature of faulting within the repository block (principal and secondary) has been adequately evaluated from the range of possible interpretations. For example, demonstrate that the DOE interpretations of trench investigations are geologically

consistent with the range of viable tectonic models and with interpretations of the crustal conditions at Yucca Mountain [SD&S 1.2.3] (NRC 1998b).

- Show that models of fault geometry, kinematics, and mechanical behavior are consistent with existing geological and geophysical results, stress and strain considerations, and viable tectonic and structural models. For example, demonstrate that projections of faults to repository depth are compatible with data from seismic reflection surveys, borehole intersections, and structural theory [SD&S 1.2.4] (NRC 1998b).

- Show that recurrence relationships for faulting are adequately derived from paleoseismic or historical earthquake data and are consistent with recurrence models used to evaluate seismicity. For example, demonstrate that a record of long recurrence intervals between large-magnitude earthquakes (determined from trenching studies) is consistent with a finding of long recurrence of large displacements [SD&S 1.2.5] (NRC 1998b).

- Show that seismic sources have been determined well enough to describe the potential sources of seismicity that will affect calculation of peak and spectral ground motions for the lifetime of the repository. For example, demonstrate that determination of the seismic sources has included: adequate characterization of the geological and tectonic setting of the site and the region; enumeration of regional earthquakes in the available historic seismic record; adequate evaluation of faults in the region, including correlation of earthquake activity with geologic structures or tectonic provinces; and adequate estimation of the earthquakes' magnitude ranges and the maximum vibratory ground motion anticipated at the site [SD&S 2.3] (NRC 1998b).

- Show that descriptions of seismic activity and recurrence relationships of fault and tectonic sources are adequate to determine ground motion at Yucca Mountain. For example, determination of the seismic activity and recurrence rates included: adequate characterization of the seismic activity rate for each source (areal or fault); adequate determination of whether seismic activity, especially maximum earthquakes, was temporally independent or occurred in clustered events; and development of an adequate recurrence rate-magnitude model for each source [SD&S 2.4] (NRC 1998b).

- Show that ground motion attenuation estimates have been determined to adequately estimate vibratory ground motions at the site. For example, demonstrate that the determination of ground motion includes adequate knowledge of site characteristics (e.g., amplification and shear wave velocities) considering uncertainties in site-specific geotechnical properties and adequate characterization of ground motion uncertainty [SD&S 2.5] (NRC 1998b).

3.2.13 Natural Resources

State the purpose of this section of Chapter 3, which is to summarize the potential for exploitable natural resources to exist at the site. Summarize the natural resources potential of the site and the surrounding region. Identify natural resource commodities and the types of deposits that have been found in environments analogous to the geologic setting. Describe the data on which the evaluation of natural resources was based. Discuss the potential for metallic mineral, industrial

rock and mineral, hydrocarbon, and geothermal resources at the site. Provide cross-references to discussions of water resources in the hydrology section.

3.2.14 Summary

State the purpose of this section of Chapter 3, which is to briefly summarize the principal geologic FEPs that could affect repository design, construction, operation, or postclosure performance. Summarize the principal geologic FEPs that are expected to favorably or unfavorably affect repository design, construction, operation, or postclosure performance. Provide cross-references to other parts of Section 3.2 that discuss these geologic FEPs. Provide cross-references to sections of Chapters 4, 5, 6, 7, and 8 that discuss effects of geologic characteristics on design, construction, operation, or postclosure performance.

3.3 GEOCHEMISTRY

State the purpose of this section of Chapter 3, which is to summarize the geochemistry of Yucca Mountain and the surrounding region. Summarize the geochemistry of Yucca Mountain and the surrounding region. Identify the topics to be discussed in this section and briefly discuss the organization of the section. Describe the geochemistry testing program, and discuss why the program was adequate to support the characterization of the geochemical system and why it was adequate to provide information needed for design and PA. Ensure all applicable requirements, as identified in the appropriate compliance program guidance package, are addressed.

Explain that Chapter 8 will describe the measures used to support the models and abstractions in compliance with proposed 10 CFR 63.114(g) (64 FR 8640).

3.3.1 Overview

State the purpose of this section of Chapter 3, which is to summarize the rock and fluid geochemistry, and the geochemical framework of the site. Summarize the general characteristics of rock geochemistry, fluid geochemistry, and the geochemical framework of the site. Discuss how the geochemistry information is used to characterize hydrology and how rock and fluid geochemistry is expected to impact repository design and postclosure performance. Identify geochemical features and processes that are expected to act as barriers (proposed 10 CFR 63.114(i) and proposed 10 CFR 63.114(j) [64 FR 8640]) and retard the migration of radionuclides. Provide cross-references to Chapter 8 for discussions of barriers. Summarize alternate interpretations of the geochemical system, and compare them in terms of their impact on the conceptual framework for the site. Briefly summarize how characteristics of the principal geochemical features and processes of the site are expected to affect the evaluation of the safety of the site. Provide cross-references to discussions in Chapter 8 of the response of the geochemical system to the design thermal load. State that the information in this section demonstrates compliance with the requirement that the SAR include "information regarding the ...geochemistry of the site" (proposed 10 CFR 63.21(c)(1)(ii) [64 FR 8640]).

3.3.2 Data Sources and Quality

State the purpose of this section of Chapter 3, which is to describe the sources and QA status of data that support the discussions of the geochemistry of Yucca Mountain and its vicinity. State that site geochemical data used as the basis for the repository safety case were collected and documented, or qualified in accordance with the QA program described in Section 1.5 of this LA. Discuss the variability, uncertainty, and propagation of errors in the data and information. Discuss the use of data from non-YMP sources. If non-YMP data are considered "accepted data" as defined in the QARD (DOE 1998, Glossary), describe the basis for considering the data "accepted." Also, summarize the use of expert elicitation, if used (proposed 10 CF 63.21(c)(10) [64 FR 8640]), and discuss the extent to which expert elicitations were conducted in accordance with the guidance in NUREG-1563 (NRC 1996) or other accepted approaches. In these discussions, emphasize that expert elicitations were conducted in conformance with the QARD (DOE 1998).

Describe the representativeness of data provided by the geochemistry program, and explain that Chapter 8 will describe the measures used to support the models and abstractions in compliance with proposed 10 CFR 63.114(g) (64 FR 8640).

3.3.3 Rock Geochemistry of Yucca Mountain and Vicinity

State the purpose of this section of Chapter 3, which is to summarize the rock geochemistry of Yucca Mountain and the surrounding region. Summarize the geochemical characteristics of Yucca Mountain rocks. Summarize the testing done to determine the distribution of primary and secondary minerals and glass in Yucca Mountain tuffs. Describe the nature and origin of primary and secondary geochemical variability. Discuss both syngenetic and diagenetic geochemical processes. Evaluate data representativeness, the differences in scale between laboratory and field data sources, the effects of data extrapolation, the effects of varying natural conditions (as appropriate), and uncertainties associated with the interpretation of data.

Provide a discussion of the links between rock and fluid geochemistry, such as fracture and matrix interactions with fluids. As appropriate, discuss the use of rock geochemistry data in evaluations of site hydrology, repository design, and repository system performance.

3.3.4 Fluid Geochemistry of Yucca Mountain and Vicinity

State the purpose of this section of Chapter 3, which is to summarize the fluid geochemistry of Yucca Mountain and the surrounding region. Summarize the properties, characteristics, and chemistry of SZ and UZ groundwater, perched water, and gas at the site and in the surrounding region, as appropriate. Include descriptions and analyses of groundwater chemistry, major and minor inorganic element content, trace element content, stable and radioactive isotopes, organic content, dissolved gas and pore gas content, particulates, colloids, temperature, and pressure. Include information on gas chemistry, such as descriptions and analyses of inorganic and organic content, stable and radioactive isotopes, aerosols, temperature, and pressure. Summarize the effects of mineral-water interactions on water composition. Discuss and evaluate differences between the geochemistry of the SZ and UZ groundwater at the site.

Discuss conceptual models for the geochemical and isotopic evolution of fluids. For fluid geochemistry models, discuss the effects of differences in scale between laboratory and field data sources, the effects of varying conditions, and uncertainties associated with the extrapolation of data and information. Discuss conceptualizations and the documentation and validation of codes used to model SZ and UZ groundwater geochemistry at the site with respect to uncertainties related to thermodynamic databases, the applicability of specific models, the appropriateness of the assumptions used to model SZ and UZ groundwater geochemistry, and the sensitivity of model results to the uncertainty of the geochemical input data.

3.3.5 Summary of the Geochemical Framework

State the purpose of this section of Chapter 3, which is to summarize the geochemical framework of Yucca Mountain. Provide a summary-level discussion of the geochemical framework of Yucca Mountain. Discuss geochemical conditions and processes that exist and have existed during the Quaternary Period (with emphasis on those that differ significantly from present conditions), and provide assessments of changes that might reasonably be expected to occur in the future. Specify the basis for expected future changes.

3.4 CLIMATOLOGY AND METEOROLOGY

State the purpose of this section of Chapter 3, which is to summarize the climatological and meteorological characteristics of the site. Identify the topics to be discussed in this section, and briefly discuss the organization of the section. Describe the climatology and meteorology testing program, and discuss why the program was adequate to support the characterization of the climatology and meteorology and the program was adequate to provide information needed for design and PA. Ensure all applicable requirements, as identified in the appropriate compliance program guidance package, are addressed.

Explain that Chapter 8 will describe the measures used to support the models and abstractions in compliance with proposed 10 CFR 63.114(g) (64 FR 8640).

3.4.1 Overview

State the purpose of this section of Chapter 3, which is to describe site climatological and meteorological conditions and to summarize how information about the climatological and meteorological systems supports evaluation of repository performance. Summarize Quaternary Period and historical climates and climate evolution. Summarize projected future climate conditions in the region surrounding the site. Discuss the extent to which data from the areas studied are representative of the climatological and meteorological conditions throughout the region. Summarize alternate interpretations of the climatological and meteorological systems, and compare them in terms of their impact on the conceptual framework for the site. Briefly summarize how the climatological and meteorological characteristics are expected to affect the evaluation of the safety of the site.

State that the information in this section complies with the requirement that the SAR contain "information regarding the...climatology and meteorology of the site" (proposed 10 CFR 63.21(c)(1)(iii) [64 FR 8640]).

3.4.2 Data Sources and Quality

State the purpose of this section of Chapter 3, which is to describe the sources of data and the QA status of data that support the discussions of the climatology, meteorology, and future climate variations of Yucca Mountain and its vicinity. State that climatological and meteorological data used as the basis for the repository safety case were collected and documented, or qualified in accordance with the QA program described in Section 1.5 of this LA [USFIC 2.9] (NRC 1998d). Discuss the variability, uncertainty, and propagation of errors in the data and information. Discuss the use of data from non-YMP sources. If non-YMP data are considered "accepted data" as defined in the QARD (DOE 1998, Glossary), describe the basis for considering the data "accepted." Also, summarize the use of expert elicitation, if used (proposed 10 CFR 63.21(c)(10) [64 FR 8640]), and discuss the extent to which expert elicitations were conducted in accordance with the guidance in NUREG-1563 (NRC 1996) or other accepted approaches [USFIC 2.5] (NRC 1998d). In these discussions, emphasize that expert elicitations were conducted in conformance with the QARD (DOE 1998).

Describe the representativeness of data provided by the climatology and meteorology testing program, and explain that Chapter 8 will describe the measures used to support the models and abstractions in compliance with proposed 10 CFR 63.114(g) (64 FR 8640).

3.4.2.1 Meteorological Monitoring Network

State the purpose of this section of Chapter 3, which is to summarize the meteorological monitoring network for the site. Briefly describe the monitoring network used to collect baseline meteorology information about the site. Describe the use of data from offsite sources, including instrument type and location and method and frequency of measurements. Describe and justify the methods used to evaluate data representativeness and adequacy.

3.4.2.2 Non-Yucca Mountain Site Characterization Project Sources

State the purpose of this section of Chapter 3, which is to summarize how climatological and meteorological information from non-YMP sources have been used to characterize site conditions. Discuss how information from non-YMP sources has been used to characterize site meteorological and climatological conditions. These sources could include National Oceanic and Atmospheric Administration facilities, such as the National Climatic Data Center and National Weather Service stations; other government facilities, such as military stations; and private organizations, such as universities, that have collected relevant and technically adequate data for study. Discuss the representativeness of data from non-YMP sources. If data from non-YMP sources are considered "accepted data" (DOE 1998, Glossary), discuss the basis for considering such data "accepted."

3.4.3 Present Climate and Meteorology

State the purpose of this section of Chapter 3, which is to describe present climate and meteorology at and around the site. Summarize the historical climatological and meteorological baseline conditions at the site to establish a basis for comparisons with past conditions and projections of foreseeable future conditions. Briefly discuss relationships between air

temperatures, regional precipitation, and the water balance of the area. Briefly discuss how meteorological information presented in this section provides input for the preclosure radiation safety assessment in Chapter 7.

3.4.3.1 Planetary and Synoptic Scale Atmospheric Features–Modern Climate

State the purpose of this section of Chapter 3, which is to describe the planetary and synoptical scale atmospheric features that control the modern climate at Yucca Mountain. Summarize the regional and local climates with regard to types of air masses, synoptic features, frontal systems, airflow patterns, and relationships between synoptic-scale atmospheric processes and site meteorological conditions. Briefly describe climatological characteristics attributable to the terrain.

Discuss whether results of DOE investigations conflict with published results from NRC staff investigations or other independent climate studies. Explain the basis for any conflicts.

3.4.3.2 Regional and Site Climatology and Meteorology

State the purpose of this section of Chapter 3, which is to characterize current climatological and meteorological conditions at and around the site. Summarize average and extreme meteorological conditions of the site, including temperature, humidity, average and extreme duration and intensity of precipitation, and average wind vectors. Briefly discuss atmospheric dispersion processes at and near the site, such as airflow trajectories, atmospheric stability conditions, depletion, and deposition characteristics. Summarize the methods used to determine atmospheric dispersion characteristics.

Summarize and provide cross-references to information on atmospheric dispersion in the preclosure radiation safety analysis in Chapter 7.

3.4.4 Paleoclimatology

State the purpose of this section of Chapter 3, which is to characterize the Quaternary paleoclimatological conditions at the site. Summarize the Quaternary climatic features and processes of the site, including atmospheric, hydrospheric, and cryospheric aspects of the successive climatic regimes. Briefly describe the methods used to determine the magnitude of past climatic changes and the rates at which those changes occurred. Identify changes in precipitation regimes and windflow patterns. Identify geological, biological, and ecological evidence that supports the analysis.

3.4.4.1 Quaternary Climate Change–Global

State the purpose of this section of Chapter 3, which is to identify and briefly describe the global factors that controlled climate change during the Quaternary Period. Summarize and discuss the key factors believed to cause global climate change on millennial time scales. Briefly discuss the correlation between the orbital parameters of Earth and the advance and decline of glacial periods, which are believed to be the primary phenomena affecting changes in precipitation and

temperature at Yucca Mountain. Describe the relationship between current meteorological characteristics and intermediate-, century-, and millennial-scale climate properties.

3.4.4.2 Regional Records of Quaternary Climate Change

State the purpose of this section of Chapter 3, which is to describe the geological records from sites throughout the region that have been used to determine the characteristics of climate change during the Quaternary Period. Summarize the characteristics of Quaternary climate change, using long-term regional records to establish the rates and timing of climate change and to link those changes to global causal factors (see Section 3.4.4.1). Summarize analyses and interpretations of geological, paleontological, and isotopic data that relate to climate change. Describe how analyses of data from all sources provide information to estimate the magnitudes and rate of changes in precipitation and air temperature related to Quaternary climate change in the southern Great Basin. Include brief discussions of data from well-studied Pleistocene lakes and remnants, such as Lake Lahonton, Mono Lake, China Lake, Owens Lake Basin, Tule Lake Basin, Death Valley, Devils Hole, and Sevier Basin, as appropriate.

3.4.4.3 Site Records of Climate Change

State the purpose of this section of Chapter 3, which is to describe the local records that have been used to characterize climate change at the site during the Quaternary Period. Summarize climate-linked hydrological records from the Yucca Mountain site. Focus the discussion on records from stable isotopes and, to a lesser extent, radiogenic isotopes. Analysis will involve linking, when possible, the isotopic data describing the nature of percolation from which calcite precipitated in Yucca Mountain to surface vegetation and air mass sources above Yucca Mountain. Summarize the relationship of records of isotopic data to records of spatial and temporal local climate change.

3.4.5 Future Climate Variation

State the purpose of this section of Chapter 3, which is to describe the expected future climate at the site. Explain that discussion of future climate variation is necessary to provide the basis for estimates of shallow infiltration, UZ percolation, and SZ flow for repository design and postclosure PA.

Discuss the use of paleoclimate information as the basis for projections of future climate variation. Note in the discussion that the NRC has specified (Stablein 1997, p. 2) that estimates of future climate variation can be based on paleoclimate information and that climate modeling is not necessary to estimate future climate variation [USFIC 2.3] (NRC 1998d). Summarize the consistency of long-term projections of climate change with known patterns of climatic cycles during the Quaternary Period, especially the last 500,000 years (Interim Guidance Section 114(k) [Dyer and Horton 1999]). Summarize the likely range of future climates at the site during the postclosure performance period. Include the results of the expert elicitation sponsored by the NRC on future climate (DeWispelare et al. 1993). Show that climate projections are based primarily on paleoclimate data, and describe why they are acceptable for use in assessment of the performance of the Yucca Mountain site. Show that the DOE has made a reasonably complete search of paleoclimate data that are available for the Yucca Mountain site and southern Great

Basin region and has adequately documented the results of that search. Show that, at a minimum, the DOE has considered information contained in Forester et al. (1996), Winograd et al (1992), Szabo et al (1994) and other reports that may become available [USFIC 2.1] (NRC 1998d).

Also show the DOE projections of long-term climate change are consistent with evidence from paleoclimate data. Specifically, show that the DOE has evaluated long-term climate change based on known patterns of climatic cycles during the Quaternary Period, especially the last 500,000 years [USFIC 2.2] (NRC 1998d).

Discuss the basis for assumptions that the present-day climatic conditions will or will not persist unchanged for 10,000 years or more [USFIC 2.9] (NRC 1998d).

Provide pointer discussions and cross-references to discussions in Section 3.5.6.2 of the hydrologic effects of climate change in the UZ (e.g., changes in infiltration, percolation, and ambient seepage) and the SZ (e.g., recharge, discharge, and flow). Provide cross-references to additional discussions in the hydrology sections of Chapter 8.

3.4.6 Summary

State the purpose of this section of Chapter 3, which is to summarize the climatological and meteorological conditions at the site. Summarize the current meteorological and climatological conditions at the site. Describe how this information may be used in preclosure radiation safety evaluations and postclosure PA. Provide cross-references to appropriate discussions in Chapters 7 and 8. Summarize the paleoclimate conditions at the site and in the southern Great Basin region, and briefly describe how those results are incorporated into estimates of future climate change. Summarize the likely range of future climates at the site during the postclosure performance period. Provide cross-references to discussions of the effects of climate change on hydrologic conditions in the hydrology section (Section 3.5.6.1).

3.5 HYDROLOGY

State the purpose of this section of Chapter 3, which is to summarize the hydrologic characteristics of the UZ and SZ, and discuss how future climate changes may affect UZ and SZ hydrologic conditions. Summarize the purpose of characterizing the hydrology of Yucca Mountain and the surrounding region. Identify the topics to be discussed in this section and briefly discuss the organization of the section. Describe the hydrology-testing program, discuss why the program was adequate to support the characterization of the hydrologic system, and provide information needed for design and PA.

Ensure all applicable requirements, as identified in the appropriate compliance program guidance package, are addressed.

Explain that Chapter 8 will describe the measures used to support the models and abstractions in compliance with proposed 10 CFR 63.114(g) (64 FR 8640).

3.5.1 Overview

State the purpose of this section of Chapter 3, which is to summarize the general features of the regional groundwater flow system. Summarize the main elements of the Death Valley drainage basin and the Death Valley regional groundwater flow system. Relate current aridity to the general lack of perennial streams, the great depth of the water table, and long groundwater flow paths. Discuss general hydrologic effects of past wetter climates and the potential for future climate changes within the regulatory compliance period. Summarize the hydrologic properties of the site and the surrounding area expected to materially affect design or performance of the repository system. Identify the hydrologic properties expected to act as barriers to radionuclide transport (proposed 10 CFR 63.114(i) and proposed 10 CFR 63.114(j) [64 FR 8640]), and provide cross-references to discussions in Chapter 8 of barriers. Provide cross-references to detailed discussions elsewhere in this section. Summarize alternate interpretations of the hydrological system and compare them in terms of their impact on the conceptual framework for the site. Briefly summarize how characteristics of the principal hydrologic features and processes of the site are expected to affect the evaluation of the safety of the site.

State that the information in this section demonstrates compliance with the regulatory requirement that the SAR shall include "information regarding the...hydrology...of the site" (proposed 10 CFR 63.21(c)(1)(ii) [64 FR 8640]).

3.5.2 Data Sources and Quality

State the purpose of this section of Chapter 3, which is to describe the sources of data and the QA status of data that support the discussions of hydrology. State that site hydrological data used as the basis for the repository safety case were collected and documented, or that they were qualified in accordance with the QA program described in Section 1.5 of this LA [USFIC 2.9] [USFIC 3.6] [USFIC 4.6] [USFIC 5.11] [USFIC 6.4] (NRC 1998d). Discuss variability, uncertainty, and the propagation of errors in the data and information. Discuss the use of data from non-YMP sources. If non-YMP data are considered "accepted data" as defined in the QARD (DOE 1998, Glossary), describe the basis for considering the data "accepted." Also, summarize the use of expert elicitation, if used (proposed 10 CFR 63.21(c)(10) [64 FR 8640]), and discuss the extent to which expert elicitations were conducted in accordance with the guidance in NUREG-1563 (NRC 1996) or other accepted approaches [USFIC 2.5] [USFIC 3.5] [USFIC 4.5] [USFIC 5.10] [USFIC 6.3] (NRC 1998d). In these discussions, emphasize that expert elicitations were conducted in conformance with the QARD (DOE 1998).

Describe the representativeness of data provided by the hydrology testing program, and explain that Chapter 8 will describe the measures used to support the models and abstractions in compliance with proposed 10 CFR 63.114(g) (64 FR 8640).

3.5.3 Regional Hydrologic Setting

State the purpose of this section of Chapter 3, which is to summarize the main characteristics of the regional hydrologic framework. Describe the surface drainage system tributary to Death Valley and how it relates to standard hydrographic areas. Describe the general distribution of precipitation and areas of recharge and discharge, occurrence of groundwater, springs, wet

playas, and consumption of water by evapotranspiration. Describe Death Valley as a regional sump. Discuss the areas and scope of groundwater resources and development. Describe the general features of the regional water budget.

3.5.4 Surface Water Hydrology

State the purpose of this section of Chapter 3, which is to describe the characteristics of the surface water hydrology system. Describe the stream network in the vicinity of Yucca Mountain and the existence of perennial surface ponds (reservoirs) in Amargosa Valley. Describe the approach to assessment of stream flows and the stream monitoring program, including collection of stream water-quality data. Discuss the potential for groundwater recharge by infiltration of periodic streamflows. Describe stream flow characteristics, including erosion and deposition. Describe the flood history and flood hazards in the vicinity of the potential repository and assessments of flood frequency and inundation potential. Relate flood potential to proposed YMP surface-facility construction and other modifications in the drainage area, such as water control structures and diversions. Explain that the ephemeral, infrequent character of streamflow precludes economic development of surface water resources. State that the information in this subsection demonstrates compliance with the regulatory requirement that the SAR shall include "information regarding surface water hydrology" (proposed 10 CFR 63.21(c)(1)(iii) [64 FR 8640]).

3.5.5 Groundwater Hydrology

State the purpose of this section of Chapter 3, which is to summarize the principal features of the UZ and SZ at Yucca Mountain. Explain that underground water at the repository site includes water in both the UZ above the water table (the top of the zone of complete saturation) and in the SZ. Discuss the location of boreholes and the distribution of data from both the UZ and SZ. Describe the general flow of water in these zones, mainly vertically downward in the UZ and laterally in the SZ. Explain the occurrence and significance of bodies of perched water in the UZ from the perspective of fluid movement in and through the UZ.

3.5.5.1 Unsaturated Zone Hydrology

State the purpose of this section of Chapter 3, which is to describe the occurrence and movement of water in the UZ. Describe the investigative approach for assessing properties of the UZ. Describe the conceptual model for water, heat, and gas flows in the UZ. Present a general introduction of infiltration of precipitation, net infiltration, percolation, occurrence of perched water and lateral flow, and flux to the water table.

Site Infiltration–State the purpose of this section of Chapter 3, which is to describe the amount, and spatial and temporal distribution of infiltration at the site. Discuss spatial and temporal variability of infiltration at Yucca Mountain, and present estimates of infiltration at the repository and the site vicinity. Discuss the modeling of flow processes and the sources and magnitude of uncertainty in estimates of fluxes. Describe alternate methods of computing recharge, and compare results of various methods. If not addressed elsewhere in the hydrology section, provide a discussion of how the information in this section addresses the following aspects of shallow infiltration in the UZ [USFIC 3] (NRC 1998d).

Hydrogeologic Properties–State the purpose of this section of Chapter 3, which is to summarize the hydrogeologic properties of the UZ. Describe the use of core testing, geophysical logging, thermal and pneumatic data, and geochemical data to develop the conceptual model of fluid flow in the UZ. Describe relative rates of flow through pore matrix, fractures, and faults. Note the isotopic evidence of nuclear-era recharge circulation to and below the repository zone. Discuss natural processes that have affected or indicated flow in the UZ, including (1) zeolitization of volcanic glass, (2) precipitation of calcite and opal on footwall fracture surfaces and bottoms of lithophysal cavities, which indicate gravity-driven flow in open fractures, and (3) indications of dehydration of zeolites and vitrophyre glass, which could have released water and thereby have affected fluid flow [ENFE 1.11] (NRC 1998e). Discuss how information in this section addresses UZ or other hydrologic model uncertainty associated with evaluating coupled thermal-hydrologic-chemical effects on seepage and flow [ENFE 1.2] (NRC 1998e).

Fluid Movement–State the purpose of this section of Chapter 3, which is to summarize the fluid flow characteristics of the UZ. Describe the movement of water and gas in the UZ and summarize the testing and monitoring of fluid movement. Discuss the implications of fluid flow in the UZ with respect to water flux in the SZ.

Perched Groundwater–State the purpose of this section of Chapter 3, which is to describe occurrences and distribution of perched water at the site. Describe the testing for evidence of perched water bodies and whether they represent a single continuous zone versus isolated separate water bodies. Discuss the implications of the occurrence of perched water as related to vertical and lateral water fluxes in the UZ.

Hydrochemical and Isotopic Evidence of Fluid Flow–State the purpose of this section of Chapter 3, which is to summarize the hydrochemical and isotopic evidence of fluid flow in the UZ. Summarize and provide cross-references to discussions in the geochemistry section as to how inorganic and isotopic chemistry provide evidence of fluid flows in the UZ. Include discussions of major ion and trace element chemistry, radioactive isotopes as indicators of time of recharge, and stable isotopes as indicators of past environmental conditions.

Conceptual Model of Fluid Flow in the Unsaturated Zone–State the purpose of this section of Chapter 3, which is to describe the conceptual model of fluid flow in the UZ. Integrate the preceding sections in a summary presentation of the conceptual model of fluid flow. Discuss how infiltration, percolation flux, capillary barriers, lateral diversion, effects of perched water on fluid flow, discrete pathways, major faults and their effect on flow, and fracture flow and fracture-matrix interactions are captured and expressed in the conceptual model [SD&S 3.6] (NRC 1998b). Discuss modifications to fractures by dissolution, precipitation, wall rock deformation, and other fracture-filling processes that would affect fracture absorption of surface water; the role of fractures as barriers to flow; isolation of fracture water from the host rock because of armoring of wall rock surface by fracture coatings; and dissolution along fractures that might enhance hydraulic conductivity [SD&S 3.5] (NRC 1998b). If not already discussed, include descriptions of features and processes that tend to retard fluid flow in the UZ.

Numerical Modeling of Fluid Flow in the Unsaturated Zone–State the purpose of this section of Chapter 3, which is to describe the quantitative model of fluid flow in the UZ. Describe how the available data from the UZ are integrated into a single comprehensive 3-D model of fluid

flow in the UZ that quantifies the flow of moisture, heat, and gas, and provides PA and repository design with a credible model of relevant UZ flow processes [USFIC 3] (NRC 1998d). Describe how the UZ model provides input for the spatial and temporal percolation flux at the repository horizon, components of matrix flow in and below the repository horizon, and probable flow paths and rates from the repository to the SZ [USFIC 4] (NRC 1998d).

Summarize how the information in this section addresses aspects of present and future deep percolation at Yucca Mountain [USFIC 4] (NRC 1998d).

Demonstrate that deep percolation has been estimated using a reasonable upper bound based on available data, or provide cross-references to a demonstration in the TSPA (Chapter 8), and associated sensitivity analyses, that further refinement of the estimate will not significantly alter the estimate of total system performance [USFIC 4.1] (NRC 1998d).

Show that estimates of future percolation provide a reasonable basis for assumed long-term average net infiltration and percolation flux. In arriving at spatial- and temporal-average values, show that variability is appropriately considered, that model parameters are averaged over appropriate spatial and temporal scales, and that the abstracted model is tested against more detailed models and field observations. If appropriate, discuss the sufficiency of using a vertical 1-D model (capable of considering heterogeneities and time-varying boundary conditions at the ground surface for calculations above the repository) versus a vertically oriented, 2-D model or a 3-D model below the repository [USFIC 4.2] (NRC 1998d).

If appropriate, discuss the basis supporting the assumption that all deep percolation below the repository level bypasses the bulk of the units of the Calico Hills nonwelded (CHn) hydrogeological unit by lateral movement above the units or by vertical flow through fractures and faults. Discuss the technical bases for any deep percolation considered to flow vertically through the matrix of the nonwelded zone. Show that such technical bases consider spatial and temporal variability and the scales at which model parameters have been averaged [USFIC 4.4] (NRC 1998d).

3.5.5.2 Saturated Zone Hydrology

State the purpose of this section of Chapter 3, which is to describe the occurrence and movement of water in the SZ. Describe the general information base for the Death Valley regional flow system and the investigative approach for assessing the flow in the SZ. Describe the conceptual models of groundwater flow in the region and the Yucca Mountain vicinity. Present a general introduction to recharge, groundwater flow, and discharge in the Alkali Flat-Furnace Creek subbasin.

Regional Groundwater Flow System–State the purpose of this section of Chapter 3, which is to describe the components of the regional groundwater flow system and the movement of groundwater within the region. Describe the Death Valley regional groundwater flow system and its subdivision into subregions and subbasins. Discuss the movement of groundwater across the boundaries of flow system, subregions, and subbasins.

Hydrogeologic Units and Properties–State the purpose of this section of Chapter 3, which is to identify the regional hydrogeologic units and describe their properties. Describe hydrogeologic units at regional scale and their properties, and relate these units to geologic and hydrogeologic units and properties as used at the Yucca Mountain site. Use correlation tables as appropriate to define relationships.

Recharge, Discharge, and Water Balances–State the purpose of this section of Chapter 3, which is to describe the spatial and temporal patterns of recharge and discharge within the regional groundwater flow system. Describe patterns of recharge and discharge within the Death Valley regional groundwater flow system and the approaches and methods for water-balance accounting. Discuss data limitations.

Potentiometric Surface–State the purpose of this section of Chapter 3, which is to describe the characteristics of the regional potentiometric surface in the vicinity of Yucca Mountain. Describe the principal features of the regional potentiometric surface and the causes of those features. Describe the rationale for treating the potentiometric surface as a single unconfined water table. Describe the occurrence of perched water and confinement. Describe regional gradients with special emphasis on extensive steep gradients and pronounced troughs. Discuss vertical differences in head where data exist. Discuss the limitations of the database as these limitations affect mapping of the potentiometric surface.

Water Resources Development–State the purpose of this section of Chapter 3, which is to summarize the development and use of groundwater in the region. Describe development of groundwater supplies for irrigation, mining, domestic, municipal, commercial, and wildlife uses within the Death Valley regional groundwater flow system, and explain that surface water resource development is uneconomic because of small and infrequent flows. Explain that legal and institutional constraints tend to constrain groundwater development to that which is readily available. Describe the present and projected water use for the YMP as related to occurrence, movement, and use of groundwater in Jackass Flats and elsewhere on the Nevada Test Site.

Regional Flow Models–State the purpose of this section of Chapter 3, which is to summarize the conceptual and numerical models of regional groundwater flow. Describe conceptual and numerical models of groundwater flow in the Death Valley regional groundwater flow system and the Nevada Test Site, and relate these modeling efforts to scale models of the SZ flow system at and downgradient from the proposed repository.

If it has not been addressed elsewhere in the hydrology section, provide a demonstration that the information in this section addresses the following aspects of SZ ambient flow conditions and dilution processes [USFIC 5] (NRC 1998d):

- Describe the extent to which conceptual flow and data uncertainties have been considered. Discuss whether uncertainties because of sparse data in some areas or low confidence in the data interpretations (e.g., see discussions in Luckey et al. 1996, pp. 51-60; Czarnecki et al. 1997, pp. 96-105) have been considered by analyzing reasonable conceptual flow alternatives supported by site data or by demonstrating through sensitivity studies that the uncertainties have little impact on repository performance [USFIC 5.1] (NRC 1998d).

- Discuss whether the DOE has used mathematical groundwater models that incorporate site-specific climatic and subsurface information. Discuss how the models were calibrated and how the physical system is represented. Discuss whether the fitted aquifer parameters compare reasonably well with observed site data, and implicitly or explicitly simulated fracturing and faulting are consistent with the data in the 3-D geologic framework model [USFIC 5.6] (NRC 1998d).

Site-Scale Saturated Zone Flow System–State the purpose of this section of Chapter 3, which is to summarize the site-scale SZ flow system. Describe the site-scale SZ flow and the features that control site SZ flow. Describe the information base for the site-scale SZ flow system, the investigative approach for assessing the flow in the SZ. Explain what information is available and what information is lacking about groundwater flow and contaminant transport.

Hydrogeologic Units and Properties–State the purpose of this section of Chapter 3, which is to identify the hydrogeologic units of the site and summarize their hydrologic properties. Describe the hydrogeologic units at site scale and their properties. Relate these units to the stratigraphic and thermal-mechanical units used in the YMP using correlation tables, as appropriate. Discuss the issue of the degree to which the groundwater flow is strata bound, and discuss the results of the testing at the C-well complex in this context.

Potentiometric Surface–State the purpose of this section of Chapter 3, which is to describe the characteristics of the potentiometric surface beneath Yucca Mountain. Describe the principal features of the site potentiometric surface and the causes of those features. Describe potentiometric surface mapping, distinguishing between large, moderate, and small hydraulic gradients. Discuss explanations for these gradients and implications regarding groundwater flow. Describe the results of the multiple well testing program and their implications with respect to conceptual and numerical modeling of SZ flow.

Vertical Hydraulic Gradients–State the purpose of this section of Chapter 3, which is to summarize the characteristics of site hydraulic gradients. Describe vertical differences in head and implications with respect to groundwater flow, with special attention to high head in the deep carbonate aquifer and the implication of this concerning recharge to the volcanic aquifers. Discuss the implication of the upward gradient at the site in the context of contaminant transport.

Potentiometric Level Fluctuations–State the purpose of this section of Chapter 3, which is to describe past fluctuations in site potentiometric levels. Describe the program of potentiometric monitoring, the range of fluctuations, and spatial and temporal variability. Discuss trend analysis and conclusions regarding causes of fluctuations.

Recharge, Discharge, and Throughflow–State the purpose of this section of Chapter 3, which is to summarize site SZ recharge, discharge, and throughflow. Discuss recharge from precipitation, potential recharge from Fortymile Canyon, and discharge toward Fortymile Wash and the Amargosa Desert. Describe the methods used to assess the quantity of throughflow beneath the site and data limitations on precision of estimates of throughflow. Discuss the adequacy of the estimates of SZ recharge, discharge, and throughflow.

3.5.6 Paleohydrology

State the purpose of this section of Chapter 3, which is to summarize the Quaternary paleohydrologic conditions at the site. Summarize the paleohydrologic characteristics of the Yucca Mountain area during the Quaternary Period and for the past 10,000 years. Discuss how local geologic conditions and plant populations determine hydrologic parameters, such as evapotranspiration, infiltration, deep percolation, groundwater seepage through a repository in the UZ, and recharge to the SZ. Discuss how changes in recharge would likely induce other changes, such as elevation of the water table and hydraulic gradient away from the repository site, and thus the rate of flow to downgradient points, far-field dilution, and land and water use.

3.5.6.1 Hydrologic Effects of Climate Change

State the purpose of this section of Chapter 3, which is to summarize the hydrologic effects of changes in Quaternary paleoclimatic conditions. Present an overview of the hydrologic effects of climatic change (NRC 1998d) since the beginning of the Quaternary Period (approximately 2,000,000 years before present), with special reference to glacial cycles over the past 500,000 years. Provide cross-references, as needed, to discussions in Section 3.4 Climatology and Meteorology.

3.5.6.2 Projected Changes in Precipitation and Temperature

State the purpose of this section of Chapter 3, which is to summarize the expected magnitudes of future climate change, and changes in temperature and precipitation, during the postclosure performance period. Describe, by reference to the Climate and Meteorology section, the expected human-influenced and natural variations in temperature and precipitation for the next 10,000 years. Relate these expected changes to hydrologic parameters, including groundwater levels at Yucca Mountain and its vicinity, recharge, discharge (e.g., spring occurrences, groundwater flow, streamflows, flooding, erosion, sediment transport), distribution of vegetation, evapotranspiration, and formation of marshes and lakes. Discuss the impacts of a return to full pluvial climate conditions for at least part of the next 10,000 years.

If not addressed elsewhere in the hydrology section, provide a discussion of how the information in this section addresses the following aspects of the hydrologic effects of climate change [USFIC 2] (NRC 1998d):

- Discuss the extent to which climate projections are based primarily on paleoclimate data, and show that the DOE has made a reasonably complete search of paleoclimate data available for the Yucca Mountain site and its vicinity, and whether the results were adequately documented. Discuss information contained in Forester et al. (1996), Winograd et al. (1992), Szabo et al. (1994), and other reports that may become available [USFIC 2.1] (NRC 1998d).

- Discuss the extent to which DOE projections of long-term climate change are consistent with evidence from paleoclimate data. Specifically, show that the DOE has evaluated long-term climate change based on known patterns of climatic cycles during the Quaternary Period, especially the last 500,000 years [USFIC 2.2] (NRC 1998d).

- Demonstrate that values for climatic parameters (e.g., times of onset of climate change, mean annual precipitation, mean annual temperature) to be used in the DOE safety case are adequately justified. Show that appropriate scientific data were used, reasonably interpreted, and appropriately synthesized into parameters such as mean annual precipitation, mean annual temperature, and long-term climate variability. Discuss, as a bounding condition, a return to full pluvial climate (higher precipitation and lower temperatures) for at least a part of the 10,000-year period [USFIC 2.4] (NRC 1998d).

3.5.6.3 Water Table Rise and Changes in Hydraulic Gradients

State the purpose of this section of Chapter 3, which is to summarize the amounts of water-table rise and changes in hydraulic gradients during the Quaternary Period, and describe the expected magnitudes of such changes during the postclosure performance period. Discuss physical and chemical evidence of previous high water tables and the effect on hydraulic gradients at Yucca Mountain and its vicinity. Project future variations for the next 10,000 years with special reference to groundwater flow rates and discharge approximately 20 km south from the underground facility (in the general location of the intersection of U.S. Route 95 and Nevada Route 373). Present quantitative estimates as feasible and describe numerical modeling of water table rise. Demonstrate that the potential future water table rise is sufficiently understood to provide assurance that the repository will not be inundated during the performance period. Provide cross-references to Chapter 6 discussions of design measures to preclude inundation of the subsurface facilities. Provide cross-references to Chapter 8 discussions of postclosure changes in the potentiometric surface that indicate the repository will not be inundated during the postclosure performance period.

- Discuss that the development of bounding values of climate-induced effects (e.g., water table rise) was based primarily on paleoclimate data. Demonstrate that an acceptably complete search of paleoclimate data pertinent to water table rise and other effects (e.g., changes in precipitation and geochemistry) of climate change available for the Yucca Mountain site and region has been made. Discuss the consideration given to information contained in Paces et al. (1996), Szabo et al. (1994), Forester et al. (1996), and other reports that may become available [USFIC 2.6] (NRC 1998d).

- Discuss the use of regional and subregional models of the SZ to predict climate-induced consequences. Discuss the calibration of these models with paleohydrology data. Demonstrate that models of the consequences of climate change are consistent with evidence from the extensive paleoclimate database. For example, discuss climate-induced water table rise expected to occur in response to elevated precipitation during future pluvial climate episodes, and demonstrate that estimates of climate-induced water table rise are consistent with the paleoclimate data. (Note: An estimate of water table rise during the late Pleistocene is 120 m (394 ft). Demonstrate that assumptions about climate-induced water table rise exceeding 10,000 years, if different from 120 m (394 ft), are adequately justified) [USFIC 2.7] (NRC 1998d).

3.5.6.4 Surface Discharges, Marshes, and Lakes

State the purpose of this section of Chapter 3, which is to summarize the effects of change in paleoclimatic conditions on surface water features and describe the effects expected during the postclosure performance period. Discuss effects of past increased effective moisture in the form of greater surface discharge to streams, wider extent of marshes, and occurrence of lakes. Project expected surficial changes for the next 10,000 years.

3.5.7 Summary

State the purpose of this section of Chapter 3, which is to summarize the hydrologic features and processes of the UZ and SZ that are expected to affect design or performance of the repository system. Also discuss relevant surface water processes and parameters, including flood potential, that are expected to affect repository design or performance. Summarize the natural features and processes of the UZ and SZ expected to act as barriers to radionuclide transport or reduce the concentrations of radionuclides.

3.6 REFERENCES

The following references were used to develop this chapter of the TGD. For the LA, this section will contain the references used to develop this chapter of the LA.

References Cited

Bell, M.J. 1996. "Issue Resolution Status Report on the Potentially Adverse Condition -- Evidence of Extreme Erosion During the Quaternary Period at Yucca Mountain." Letter from M.J. Bell (NRC) to R.A. Milner (DOE/OCRWM), February 29, 1996, with enclosure, "Status of Open Items Related to the Potentially Adverse Condition Extreme Erosion During the Quaternary Period at Yucca Mountain, Nevada." ACC: HQO.19960319.0011; HQO.19960319.0012.

CRWMS (Civilian Radioactive Waste Management System) M&O (Management and Operating Contractor) 1996. *Probabilistic Volcanic Hazard Analysis for Yucca Mountain, Nevada.* BA0000000-01717-2200-00082 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19971201.0221.

CRWMS M&O 1998. *Probabilistic Seismic Hazard Analyses for Fault Displacement and Vibratory Ground Motion at Yucca Mountain, Nevada.* Final Report. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19981207.0393.

Czarnecki, J.B.; Faunt, C.C.; Gable, G.W.; and Zyvoloski, G.A. 1997. *Site Saturated-Zone Synthesis Report.* Milestone Report SP23NM3. Preliminary Report. Denver, Colorado: U.S. Geological Survey. ACC: MOL.19980204.0511.

DOE (U.S. Department of Energy) 1988. *Site Characterization Plan: Yucca Mountain Site, Nevada Research and Development Area, Nevada.* DOE/RW-0199. Eight volumes. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: HQO.19881201.0002.

DOE 1998. *Quality Assurance Requirements and Description.* DOE/RW-0333P, Rev. 8. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.19980601.0022.

DeWispelare, A.R.; Herren, T.L.; Miklas, M.P.; and Clemen, R.T. 1993. *Expert Elicitation of Future Climate in the Yucca Mountain Vicinity, Iterative Performance Assessment Phase 2.5.* CNWRA 93-016. San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. TIC: 238416.

Dyer, J.R. and Horton, D.G. 1999. "Interim Guidance Pending Issuance of New Nuclear Regulatory Commission (NRC) Regulations for Yucca Mountain, Nevada." Letter from J.R. Dyer (DOE/YMSCO) to D.R. Wilkins (CRWMS M&O), June 18, 1999, OL&RC:AVG:1435, with enclosure, "Interim Guidance Pending Issuance of New NRC Regulations for Yucca Mountain." ACC: MOL.19990712.0039.

Forester, R.M.; Bradbury, J.P.; Carter, C.; Elvidge, A.B.; Hemphill, M.L.; Lundstrom, S.C.; Mahan, S.A.; Marshall, B.D.; Neymark, L.A.; Paces, J.B.; Sharpe, S.E.; Whelan, J.F.; and Wigand, P.E. 1996. *Synthesis of Quaternary Response of the Yucca Mountain Unsaturated and Saturated Zone Hydrology to Climate Change.* Milestone Report 3GCA102M. Denver, Colorado: U.S. Geological Survey. ACC: MOL.19970211.0026.

Luckey, R.R.; Tucci, P.; Faunt, C.C.; Ervin, E.M.; Steinkampf, W.C.; D'Agnese, F.A.; and Patterson, G.L. 1996. *Status of Understanding of the Saturated-Zone Ground-Water Flow System at Yucca Mountain, Nevada, as of 1995.* Water-Resources Investigations Report 96-4077. Denver, Colorado: U.S. Geological Survey. ACC: MOL.19970513.0209.

NRC (U.S. Nuclear Regulatory Commission) 1996. *Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program.* NUREG-1563. Washington, D.C.: U.S. Nuclear Regulatory Commission. TIC: 226832.

NRC 1998a. *Issue Resolution Status Report Key Technical Issue: Repository Design and Thermal-Mechanical Effects.* Rev. 1. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.19981130.0219.

NRC 1998b. *Issue Resolution Status Report Key Technical Issue: Structural Deformation and Seismicity.* Rev. 1. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.19981202.0550.

NRC 1998c. *Issue Resolution Status Report Key Technical Issue: Igneous Activity.* Rev. 1. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.19980909.0045.

NRC 1998d. *Issue Resolution Status Report Key Technical Issue: Unsaturated and Saturated Flow Under Isothermal Conditions.* Rev. 1. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.19990105.0142.

NRC 1998e. *Issue Resolution Status Report Key Technical Issue: Evolution of the Near-Field Environment.* Rev. 1. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.19981106.0144.

Paces, J.B.; Forester, R.M.; Whelan, J.F.; Mahan, S.A.; Bradbury, J.P.; Quade, J.; Neymark, L.A.; and Kwak, L.M. 1996. *Synthesis of Ground-Water Discharge Deposits Near Yucca Mountain.* Milestone Report 3GQH671M. Denver, Colorado: U.S. Geological Survey. ACC: MOL.19970205.0007.

Stablein, N.K. 1997. "Issue Resolution Status Report on Methods to Evaluate Climate Change and Associated Effects at Yucca Mountain (Key Technical Issue: Unsaturated and Saturated Flow Under Isothermal Conditions)." Letter from N.K. Stablein (NRC) to S.J. Brocoum (DOE/YMSCO), June 30, 1997, with enclosure, "Issue Resolution Status Report on Methods to Evaluate Climate Change and Associated Effects at Yucca Mountain (Key Technical Issue: Unsaturated and Saturated Flow Under Isothermal Conditions). ACC: MOL.19980219.0879; MOL.19980219.0880.

Szabo, B.J.; Kolesar, P.T.; Riggs, A.C.; Winograd, I.J.; and Ludwig, K.R. 1994. "Paleoclimatic Inferences from a 120,000-Yr Calcite Record of Water-Table Fluctuation in Browns Room of Devils Hole, Nevada." *Quaternary Research, 41,* 59-69. New York, New York: Academic Press. TIC: 234642.

Winograd, I.J.; Coplen, T.B.; Landwehr, J.M.; Riggs, A.C.; Ludwig, K.R.; Szabo, B.J.; Kolesar, P.T.; and Revesz, K.M. 1992. "Continuous 500,000-Year Climate Record from Vein Calcite in Devils Hole, Nevada." *Science, 258,* 255-260. Washington, D.C.: American Association for the Advancement of Science. TIC: 237563.

YMP (Yucca Mountain Site Characterization Project) 1993. *Evaluation of the Potentially Adverse Condition "Evidence of Extreme Erosion During the Quaternary Period" at Yucca Mountain, Nevada.* Topical Report. YMP/92-41-TPR. Las Vegas, Nevada: Yucca Mountain Site Characterization Office. ACC: NNA.19930316.0208.

YMP 1997. *Methodology to Assess Fault Displacement and Vibratory Ground Motion Hazards at Yucca Mountain.* Topical Report. YMP/TR-002-NP, Rev. 1. Las Vegas, NV: Yucca Mountain Site Characterization Office. ACC: MOL.19971016.0777.

Codes, Standards, and Regulations

10 CFR (Code of Federal Regulations) 60. Energy: Disposal of High-Level Radioactive Wastes in Geologic Repositories. Readily available.

64 FR (Federal Register) 8640. Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, Nevada. Proposed rule 10 CFR 63. Readily available.

Nuclear Waste Policy Act of 1982. 42 U.S.C. 10101 et seq. TIC: 241722.

CHAPTER 4. REPOSITORY SURFACE DESIGN

Chapter 4 describes the geologic repository operations area (GROA) surface facilities. Authors preparing the License Application (LA) shall use this document for guidance and must read the Introduction and Appendix B before writing their respective sections.

Proposed 10 CFR 63.24(b) (64 FR 8640) recognizes that all information may not be available at the time of docketing the LA. The guidance in this chapter adopts a differentiated approach for some required information. A differentiated approach means providing the information required at construction authorization (CA) versus that required at license to receive and possess high-level radioactive waste (HLW). Information in any section that is identified as "Information Required at the Time of Construction Authorization" will be provided at the time the LA is submitted for docketing. The information at the CA stage will be replaced or supplemented with additional information developed since docketing the LA. Some of the information that is required at the time to receive and possess may be totally new information. The additional information is identified as "Information Required at the Time of Update to the License Application to Receive and Possess." Because the Repository Surface Facility design has licensing precedent, the level of detail required in Chapter 4 will be less than is required for those chapters that contain the structures, systems, and components (SSCs) that do not have licensing precedent. The minimum required level of design detail for the Surface Facility SSCs is provided in Section 4.1.

4. PURPOSE AND SUMMARY

Provide an introductory discussion outlining the purpose and organization of this chapter. Describe the mission of the GROA. Include a mission statement, planned waste disposal totals, and emplacement duration for wastes (i.e., 24 years).

Provide a summary and conclusions regarding how the surface facility design process and specific designs support safe operation of the repository as demonstrated in Chapter 7 of the LA.

4.1 SURFACE FACILITIES DESIGN

Briefly describe how the repository is designed to withstand natural phenomena such as seismic events. Refer to Chapter 7 for the specific natural phenomena considered and evaluated for design basis events (DBEs). State that the surface SSCs design adequately addresses these DBEs, demonstrating that the performance objectives of the proposed 10 CFR 63.111 (64 FR 8640) can be met given the occurrence of these events.

Provide an overview of the major site features that affect GROA design and performance. Refer to Chapter 3 for the detailed description of this information. Provide a reference to additional descriptions of site conditions expected to be encountered in constructing the surface facilities. Discuss the site-related design bases relevant to the GROA design. The site-related design bases include ambient temperature and humidity extremes, seismic loadings, maximum wind loadings (including tornado), meteorology, and foundation design assumptions. Briefly describe the interpretation of site geology such as stratigraphy, structural features, major and minor faults, old

volcanoes, and history of seismic activity. Provide a reference to Chapter 3 for sources of site information and descriptions.

Describe the surface drainage characteristics.

Provide the following information and reference Chapter 3 for details:

- Soil properties and other relevant data for the design of foundations
- Relevant meteorological data
- Rock data and properties relevant to the design of surface facilities.

Describe the U.S. Department of Energy and Civilian Radioactive Waste Management System (CRWMS) Management and Operating Contractor (M&O) documents that direct and control the repository design such as the generic requirements documents, the system description documents (SDDs), and system and subsystem requirements documents. State that the repository design also meets any other legal requirements of federal, state, and local agencies applicable to the mined geologic disposal of spent nuclear fuel (SNF) and HLW.

Briefly discuss the integrated safety analysis (ISA) performed to ensure the preclosure performance objectives are satisfied. Refer to Chapter 7 of the LA for a detailed discussion of the ISA. Briefly discuss how the ISA results are reflected in this chapter.

4.1.1 Surface Facility Design Overview

Provide a general description of the GROA surface facilities, including surface operational and support facilities that are Quality Level (QL) 1, 2, or 3. Identify the major surface operations performed at the facility, which are related to QL 1, 2, or 3 SSCs, while considering the following: receipt of HLW, cask maintenance, development operations, waste management, and balance of plant functions. Provide a brief overview of the waste stream to be placed in the repository and a cross-reference to more detailed discussions in Chapter 5. Describe the relationship and interface with the other portions of the monitored geologic repository. Provide the location of the GROA relative to the boundaries of the site. Provide drawings or figures to identify the appropriate facilities and GROA. Provide key descriptive parameters for the facilities, such as number of facilities, construction type, site area, overall floor space, and site location and terrain.

Briefly describe or list the major design features important to safety, retrievability, and waste isolation.

4.1.2 Level of Design Detail

This section provides the guidance for the required level of detail that must be provided for the SSCs that are discussed in Chapter 4. There will not be a Section 4.1.1 in the LA.

QL 1, 2, and 3 SSCs and non-safety SSCs are required to be described in the LA per the *Level of Design Detail Necessary for the License Application for Construction Authorization* (CRWMS M&O 1999). See Section 2.4 for more information related to QL classifications. For any SSC, the level of design detail required depends on the following (CRWMS M&O 1999):

- Importance to protection of public health and safety
- Importance to protection of worker health and safety
- Need to demonstrate compliance with regulatory requirements
- Need to support submittal of a docketable LA.

The QL information following this paragraph lists the specific information to be included in the description of SSCs that fall into one of the three QL classifications. If a specific information item is not relevant for a given SSC, that information item need not be addressed for that SSC in the LA.

4.1.2.1 Quality Level 1 Structures, Systems, and Components

The following information, as applicable, must be provided for each QL 1 SSC.

Regulatory Bases–These are primarily the proposed 10 CFR 63 (64 FR 8640) requirements.

- Identify the SSCs relied upon to limit or prevent potential accidents or mitigate their consequences; refer to Chapter 7 for details on the analysis that identified the SSCs as such. Identify the specific DBEs. Describe the design features incorporated into the SSCs and describe the function of the SSCs, including controls that are relied upon to limit, prevent, or mitigate the consequences of DBEs. Use drawings such as piping and instrument diagrams, electrical one-line diagrams, general arrangement drawings, and handling drawings as necessary to present the information. Identify measures taken to ensure the availability of identified safety systems (Interim Guidance Section 112(e) [Dyer and Horton 1999]). Discuss the design considerations that prevent releases of radioactive materials that could result in a dose of 0.25 mSv (25 mrem) to an individual member of the public at the boundary. Where appropriate, identify the sequence of events and how the system responds to the event (proposed 10 CFR 63.111(b)(1) [64 FR 8640]; Interim Guidance Section 111(a)(2) and Interim Guidance Section 111(b)(2) [Dyer and Horton 1999]).

- Identify the SSCs that limit and control radiation exposures and radiation levels in restricted and unrestricted areas and the release of radioactive materials to unrestricted areas, and address the limits of 10 CFR 20.1201, 10 CFR 20.1301, and the as low as is reasonably achievable (ALARA) provisions of 10 CFR 20.1101(b) and 10 CFR 20.1101(d). The limits are as follows (proposed 10 CFR 63.111(a)(1) and proposed 10 CFR 63.111(b)(1) [64 FR 8640]; Interim Guidance Section 111(a)(2) [Dyer and Horton 1999]):

  - No worker shall receive the more limiting total effective dose equivalent (TEDE) of 0.05 Sv (5 rem) or the sum of deep-dose equivalent and committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.50 Sv (50 rem) annually. The annual dose equivalent to the lens of the eye shall not exceed 0.15 Sv (15 rem). The annual shallow dose equivalent to the skin or any extremity shall not exceed 0.50 Sv (50 rem).

- No individual member of the public shall receive a TEDE in excess of 1 mSv (0.1 rem) annually, and the dose in any unrestricted area from external sources does not exceed 0.02 mSv (0.002 rem) per hour.

- Identify ALARA design considerations for facility features that limit and control occupational dose and dose to members of the public as required by 10 CFR 20.1101(b) and 10 CFR 20.1101(d) (proposed 10 CFR 63.111(a)(1) [64 FR 8640]).

- Describe the design considerations for systems that monitor and control effluents. Describe design considerations for facility features and systems that control and monitor radiation levels to limit occupational radiation exposure (proposed 10 CFR 63.21(c)(14) and proposed 10 CFR 63.111(a)(1) [64 FR 8640]).

License Specifications

- **Information Required at the Time of Construction Authorization**–When discussing equipment or parameters credited in the safety analysis for mitigating the consequences of a Category 1 or 2 DBE, note that the item is being considered as a subject for a license specification in accordance with proposed 10 CFR 63.21(c)(13) (64 FR 8640). Include supporting information to demonstrate how and why the item is credited in the safety analysis for mitigating the consequences of a Category 1 or 2 DBE. Refer to Chapter 11 for an identification of the license specifications and summary of their justification for being considered as a license specification (proposed 10 CFR 63.21(c)(13) [64 FR 8640]).

- **Information Required at the Time of Update to the License Application to Receive and Possess**–Identify equipment and parameters used to mitigate the consequences of a Category 1 or 2 DBE as being addressed in the license specifications. Refer to Chapter 11 for a discussion of the license specification development process (proposed 10 CFR 63.21(c)(13) [64 FR 8640]).

Applicable Codes and Standards–List the codes and standards (including guidance documents and technical positions) required in the design and construction of the GROA. The codes, standards, and other documents should be listed on a structure or system level. The codes and standards listed must be the same as those found in corresponding SDDs (proposed 10 CFR 63.21(c)(3) [64 FR 8640]).

Design Criteria–A design criterion is a standard or rule against which a design can be judged. The U.S. Nuclear Regulatory Commission (NRC) defines principal design criteria in 10 CFR 50, Appendix A, as criteria that establish the necessary design, fabrication, construction, and performance requirements for SSCs important to safety (CRWMS M&O 1999). Examples include:

- The lifetime of the waste package (WP) shall be long enough to contain the waste throughout the thermal period.

- Provisions shall be made so that, if there is a loss of the primary electric power source or circuit, reliable and timely emergency power can be provided to instruments, utility service systems, and operating systems, including alarm systems, important to safety.
- The facility design shall comply with the ALARA criteria of 10 CFR 20.
- See the example table below.

Summary of Centralized Interim Storage Facility Principal Design Criteria

Design Parameter	Design Criteria	Condition	Applicable Codes, Standards, & Bases
Seismic (Ground Motion)	Design response spectra anchored at horizontal acceleration of 0.75g	Accident	N/A
Seismic (Surface Faulting)	No surface faulting	Accident	N/A

Source: DOE 1998

The design criteria are found in the appropriate SDD.

Design Bases–Design bases refer to the information that identifies the specific functions to be performed by an SSC of a facility and the specific values or range of values chosen for controlling parameters as reference bounds for design. These values may be restraints derived from generally accepted "state-of-the-art" practices for achieving functional goals or requirements derived from analysis (based on calculation or experiments) of the effects of a postulated event under which an SSC must meet its functional goals. The values for controlling parameters for external events include (CRWMS M&O 1999):

- Estimates of severe natural events, to be used for deriving design bases, that will be based on consideration of historical data on the associated parameters, physical data, or analysis of upper limits of the physical processes involved
- Estimates of severe external human-induced events, to be used for deriving design bases, that will be based on analysis of human activity in the region, taking into account the site characteristics and the risks associated with the event.

The project design bases are the combination of the system functions and the performance parameters found in the SDDs.

General Description–This is an overall description of the SSCs, equipment, and process activities. Base this description on the information contained in the related SDDs. Include in this description the information required to support the safety analysis of the system or that which can be readily derived from it. Provide a summary of the system functions, operations, the system design, concept of operations, and a description of system interfaces. Include a discussion on any special construction or fabrication techniques, unique testing programs, or special design and analysis procedures used for the SSCs, as applicable. Include diagrams to show concepts or ideas as needed to support the text (proposed 10 CFR 63.112(a) [64 FR 8640]).

Discussion of the materials of construction (including general arrangement and approximate dimensions) for the GROA (proposed 10 CFR 63.21(c)(3) [64 FR 8640]) includes the following:

- **Information Required at the Time of Construction Authorization**–Include only the principal materials used in the design of SSCs that either prevent or mitigate a Category 1 or 2 DBE or are required for worker safety. Examples of principal materials are structural such as steel beam construction or reinforced concrete, systems components such as stainless steel piping, high-efficiency particulate air (HEPA) filters with charcoal absorbers, or leaded glass used for shielding. The materials of construction should be discussed on an SSC level as they are described in the various sections.

- **Information Required at the Time of Update to the License Application to Receive and Possess**–In addition to information provided at the time of CA, present a detailed description of the materials of construction. This should be done by component or subsystem as each is discussed in the description of the specific SSCs. The information should include the compatibility of the material with its environment during normal operations or post-accident situations, whichever is the most limiting as it relates to the service that the component provides. Three examples are provided here:

  - HEPA filters

    HEPA filter elements are made of pleated fiberglass with an aluminum separator design, measure 24 x 24 x 11.5 in., and are each capable of handling a nominal flow rate of 1,000 ft^3/min. The filter medium is cased in stainless steel, has face guards on both sides, and is water- and fire-resistant.

  - Cooling coils

    The cooling coils are of nonferrous construction with aluminum fins mechanically bonded to seamless copper tubing. Coils are arranged for counter-flow operation using chilled water. The tube bundle is enclosed in a steel frame.

  - Low total dissolved solids holdup tank (T-01 C)

Quantity per unit	=	1
Capacity (each)	=	30,000 gal
Design pressure and temperature	=	Atmospheric pressure and 150°F
Operating pressure and temperature	=	Atmospheric pressure and 80°F
Material	=	304 stainless steel

Research and Development

- **Information Required at the Time of Construction Authorization**–Identify SSCs that require research and development (R&D) to confirm the adequacy of design. Provide available information that describes the type of R&D required and the reason the

additional information is needed. Refer to Chapter 11 for a description of the R&D program and the proposed schedule (proposed 10 CFR 63.21(c)(21) [64 FR 8640]).

- **Information Required at the Time of Update to the License Application to Receive and Possess**–Provide the results from the required R&D identified in the LA at the time of CA for the various SSCs. Discuss the reason the additional information was originally needed (proposed 10 CFR 63.21(c)(21) [64 FR 8640]).

Decontamination

Information Required at the Time of Update to the License Application to Receive and Possess–Provide the specific decontamination design considerations as part of the SSC discussion (proposed 10 CFR 63.21(c)(18) [64 FR 8640]).

Drawings and Diagrams–The drawings and diagrams required for QL 1 SSCs included and discussed in Chapter 4 are those that show information to support the safety case. Typical types of drawings and diagrams are listed below (proposed 10 CFR 63.112(a) [64 FR 8640]).

- Piping and instrument diagrams
- Electrical one-line diagrams
- General arrangement drawings
- Handling diagrams.

4.1.2.2 Quality Level 2 Structures, Systems, and Components

The following information, as applicable, must be provided for each QL 2 SSC.

Regulatory Bases–These are primarily the proposed 10 CFR 63 (64 FR 8640) requirements.

- Identify the SSCs relied upon to limit or prevent potential accidents or mitigate their consequences; refer to Chapter 7 for details on the analysis that identified the SSC as such. Identify the specific DBEs. Identify and describe the design features incorporated into the SSC and describe the function of the SSCs, including controls that are relied upon to limit, prevent, or mitigate the consequences of DBEs. Include identification of measures taken to ensure the availability of identified safety systems (Interim Guidance Section 112(e) [Dyer and Horton 1999]). Discuss the design considerations that prevent releases of radioactive materials that could result in a dose of 0.25 mSv (25 mrem) to an individual member of the public at the boundary. Where appropriate, identify the sequence of events and how the system responds to the event (proposed 10 CFR 63.111(b)(1) [64 FR 8640]; Interim Guidance Section 111(a)(2) and Interim Guidance Section 111(b)(2) [Dyer and Horton 1999]).

- Identify the SSCs that limit and control radiation exposures and radiation levels in restricted and unrestricted areas and the release of radioactive materials to unrestricted areas, and address the limits of 10 CFR 20.1201, 10 CFR 20.1301, and the ALARA provisions of 10 CFR 20.1101(b) and 10 CFR 20.1101(d). The limits are as follows

(proposed 10 CFR 63.111(a)(1) and proposed 10 CFR 63.111(b)(1) [64 FR 8640]; Interim Guidance Section 111(a)(2) [Dyer and Horton 1999]):

- No worker shall receive the more limiting TEDE of 0.05 Sv (5 rem), or the sum of deep-dose equivalent and committed dose equivalent, to any individual organ or tissue (other than the lens of the eye) of 0.50 Sv (50 rem) annually. The annual dose equivalent to the lens of the eye shall not exceed 0.15 Sv (15 rem). The annual shallow dose equivalent to the skin or any extremity shall not exceed 0.50 Sv (50 rem).

- No individual member of the public shall receive a TEDE in excess of 1 mSv (0.1 rem) annually, and the dose in any unrestricted area from external sources does not exceed 0.02 mSv (0.002 rem) per hour.

- Identify ALARA design considerations for facility features that limit and control occupational dose and dose to members of the public as required by 10 CFR 20.1101(b) and 10 CFR 20.1101(d) (proposed 10 CFR 63.111(a)(1) [64 FR 8640]).

- Describe the design considerations for systems that monitor and control effluents. Describe design considerations for facility features and systems that control and monitor radiation levels to limit occupational radiation exposure (proposed 10 CFR 63.21(c)(14) and proposed 10 CFR 63.111(a)(1) [64 FR 8640]).

License Specifications

- **Information Required at the Time of Construction Authorization**–When discussing equipment or parameters credited in the safety analysis for mitigating the consequences of a Category 1 or 2 DBE, note that the item is being considered as a subject for a license specification in accordance with proposed 10 CFR 63.21(c)(13) (64 FR 8640). Include the supporting information to demonstrate how and why the item is credited in the safety analysis for mitigating the consequences of a Category 1 or 2 DBE. Refer to Chapter 11 for an identification of the license specifications and summary of their justification for being considered as a license specification (proposed 10 CFR 63.21(c)(13) [64 FR 8640]).

- **Information Required at the Time of Update to the License Application to Receive and Possess**–Identify equipment and parameters used to mitigate the consequences of a Category 1 or 2 DBE as being addressed in the license specifications. Refer to Chapter 11 for a discussion of the license specification development process (proposed 10 CFR 63.21(c)(13) [64 FR 8640]).

Applicable Codes and Standards–List the codes and standards (including guidance documents and technical positions) required in the design and construction of the GROA. The codes, standards, and other documents identified should be listed on a structure or system level. The codes and standards listed must be the same as those found in corresponding SDDs (proposed 10 CFR 63.21(c)(3) [64 FR 8640]).

Design Criteria–A design criterion is a standard or rule against which a design can be judged. The NRC defines principal design criteria in 10 CFR 50, Appendix A, as criteria that establish the necessary design, fabrication, construction, and performance requirements for SSCs important to safety (CRWMS M&O 1999). Examples include the following:

- The lifetime of the WP shall be long enough to contain the waste throughout the thermal period.
- Provisions shall be made so that, if there is a loss of the primary electric power source or circuit, reliable and timely emergency power can be provided to instruments, utility service systems, and operating systems, including alarm systems, important to safety.
- The facility design shall comply with the ALARA criteria of 10 CFR 20.
- See the example table in Section 4.1.1.1.

The design criteria are found in the SDD.

General Description–This is an overall description of the SSCs, equipment, and process activities. Base this description on the information contained in the related SDDs. Include in this description the information required to support the safety analysis of the system or that which can be readily derived from it. Provide a summary of the system functions, operations, the system design, concept of operations, and a description of system interfaces. Include a discussion on any special construction or fabrication techniques, unique testing programs, or special design and analysis procedures used for the SSCs, as applicable. Include diagrams to show concepts or ideas as needed to support the text (proposed 10 CFR 63.112(a) [64 FR 8640]).

Discussion of the materials of construction (including general arrangement and approximate dimensions) for the GROA includes the following (proposed 10 CFR 63.21(c)(3) [64 FR 8640]):

- **Information Required at the Time of Construction Authorization**–Include only the principal materials used in the design of SSCs that either prevent or mitigate a Category 1 or 2 DBE or are required for worker safety. Examples of principal materials are structural such as steel beam construction or reinforced concrete, systems components such as stainless steel piping, HEPA filters with charcoal absorbers, or leaded glass used for shielding. The materials of construction should be discussed on an SSC level as they are described in the various sections.
- **Information Required at the Time of Update to the License Application to Receive and Possess**–In addition to information provided at the time of CA, a detailed description of the materials of construction must be presented. This should be done by component as each is discussed in the description of the specific SSCs. The information should include the compatibility of the material with its environment during normal operations or post-accident situations, whichever is the most limiting and as it relates to the service that the component provides. See the examples provided under QL 1 SSCs.

Research and Development

- **Information Required at the Time of Construction Authorization**–Identify SSCs that require R&D to confirm the adequacy of design. Provide available information that describes the type of R&D required and the reason the additional information is needed. Refer to Chapter 11 for a description of the R&D program and the proposed schedule (proposed 10 CFR 63.21(c)(21) [64 FR 8640]).

- **Information Required at the Time of Update to the License Application to Receive and Possess**–Provide the results from the required R&D identified in the LA at the time of CA for the various SSCs. Discuss the reason the additional information was originally needed (proposed 10 CFR 63.21(c)(21) [64 FR 8640]).

Decontamination

Information Required at the Time of Update to the License Application to Receive and Possess–Provide the specific decontamination design considerations as part of the SSC discussion (proposed 10 CFR 63.21(c)(18) [64 FR 8640]).

4.1.2.3 Quality Level 3 Structures, Systems, and Components

The information, as applicable, that must be provided for each QL 3 SSC includes the following.

Regulatory Bases–These are primarily the proposed 10 CFR 63 (64 FR 8640) requirements.

- Identify the SSCs relied upon to limit or prevent potential accidents or mitigate their consequences; refer to Chapter 7 for details on the analysis that identified the SSC as such. Identify and describe the function of the SSCs, including controls that are relied upon to limit, prevent, or mitigate the consequences of DBEs. Include identification of measures taken to ensure the availability of identified safety systems (Interim Guidance Section 112(e) [Dyer and Horton 1999]). Discuss the design considerations that prevent releases of radioactive materials that could result in a dose of 0.25 mSv (25 mrem) to an individual member of the public at the boundary. Where appropriate, identify the sequence of events and how the system responds to the event (proposed 10 CFR 63.111(b)(1) [64 FR 8640]; Interim Guidance Section 111(a)(2) and Interim Guidance Section 111(b)(2) [Dyer and Horton 1999]).

- Identify the SSCs that limit and control radiation exposures and radiation levels in restricted and unrestricted areas and the release of radioactive materials to unrestricted areas, and address the limits of 10 CFR 20.1201, 10 CFR 20.1301, and the ALARA provisions of 10 CFR 20.1101(b) and 10 CFR 20.1101(d). The limits are as follows (proposed 10 CFR 63.111(a)(1) and proposed 10 CFR 63.111(b)(1) [64 FR 8640]; Interim Guidance Section 111(a)(2) [Dyer and Horton 1999]):

  - No worker shall receive the more limiting TEDE of 0.05 Sv (5 rem), or the sum of deep-dose equivalent and committed dose equivalent, to any individual organ or tissue (other than the lens of the eye) of 0.50 Sv (50 rem) annually. The annual dose

equivalent to the lens of the eye shall not exceed 0.15 Sv (15 rem). The annual shallow dose equivalent to the skin or to any extremity shall not exceed 0.50 Sv (50 rem).

- No individual member of the public shall receive a TEDE in excess of 1 mSv (0.1 rem) annually, and the dose in any unrestricted area from external sources does not exceed 0.02 mSv (0.002 rem) per hour.

- Identify ALARA design considerations for facility features that limit and control occupational dose and dose to members of the public as required by 10 CFR 20.1101(b) and 10 CFR 20.1101(d) (proposed 10 CFR 63.111(a)(1) [64 FR 8640]).

- Describe the design considerations for systems that monitor and control effluents. Describe design considerations for facility features and systems that control and monitor radiation levels to limit occupational radiation exposure (proposed 10 CFR 63.21(c)(14) and proposed 10 CFR 63.111(a)(1) [64 FR 8640]).

License Specifications

- **Information Required at the Time of Construction Authorization**–When discussing equipment or parameters credited in the safety analysis for mitigating the consequences of a Category 1 or 2 DBE, note that the item is being considered as a subject for a license specification in accordance with proposed 10 CFR 63.21(c)(13) (64 FR 8640). Include supporting information to demonstrate how and why the item is credited in the safety analysis for mitigating the consequences of a Category 1 or 2 DBE. Refer to Chapter 11 for an identification of the license specifications and summary of their justification for being considered as a license specification (proposed 10 CFR 63.21(c)(13) [64 FR 8640]).

- **Information Required at the Time of Update to the License Application to Receive and Possess**–Identify equipment and parameters used to mitigate the consequences of a Category 1 or 2 DBE as being addressed in the license specifications. Refer to Chapter 11 for a discussion of the license specification development process (proposed 10 CFR 63.21(c)(13) [64 FR 8640]).

Design Criteria–A design criterion is a standard or rule against which a design can be judged. The NRC defines principal design criteria in 10 CFR 50, Appendix A, as criteria that establish the necessary design, fabrication, construction, and performance requirements for SSCs important to safety (CRWMS M&O 1999). Examples include the following:

- The lifetime of the WP shall be long enough to contain the waste throughout the thermal period.

- Provisions shall be made so that, if there is a loss of the primary electric power source or circuit, reliable and timely emergency power can be provided to instruments, utility service systems, and operating systems, including alarm systems, important to safety.

- The facility design shall comply with the ALARA criteria of 10 CFR 20.
- See the example table in Section 4.1.1.1.

The design criteria are found in the SDD.

General Description–This is an overall description of the SSCs, equipment, and process activities. Base this description on the information contained in the related SDDs. Include in this description the information required to support the safety analysis of the system or that which can be readily derived from it. Provide a summary of the system functions, operations, the system design, concept of operations, and a description of system interfaces. Include a discussion on any special construction or fabrication techniques, unique testing programs, or special design and analysis procedures used for the SSCs, as applicable. Include diagrams to show concepts or ideas as needed to support the text (proposed 10 CFR 63.112(a) [64 FR 8640]).

Discussion of the materials of construction (including general arrangement and approximate dimensions) for the GROA include the following (proposed 10 CFR 63.21(c)(3) [64 FR 8640]):

- **Information Required at the Time of Construction Authorization**–Include only the principal materials used in the design of SSCs that either prevent or mitigate a Category 1 or 2 DBE or are required for worker safety. Examples of principal materials are structural such as steel beam construction or reinforced concrete, systems components such as stainless steel piping, HEPA filters with charcoal absorbers, or leaded glass used for shielding. The materials of construction should be discussed on an SSC level as they are described in the various sections.
- **Information Required at the Time of Update to the License Application to Receive and Possess**–In addition to information provided at the time of CA, a detailed description of the materials of construction must be presented. This should be done by component as each is discussed in the description of the specific SSCs. The information should include the compatibility of the material with its environment during normal operations or post-accident situations, whichever is the most limiting and as it relates to the service that the component provides. See the examples provided under QL 1 SSCs.

Research and Development

- **Information Required at the Time of Construction Authorization**–Identify SSCs that require R&D to confirm the adequacy of design. Provide available information that describes the type of R&D required and the reason the additional information is needed. Refer to Chapter 11 for a description of the R&D program and the proposed schedule (proposed 10 CFR 63.21(c)(21) [64 FR 8640]).
- **Information Required at the Time of Update to the License Application to Receive and Possess**–Provide the results from the required R&D identified in the LA at the time of CA for the various SSCs. Discuss the reason the additional information was originally needed (proposed 10 CFR 63.21(c)(21) [64 FR 8640]).

Decontamination

Information Required at the Time of Update to the License Application to Receive and Possess–Provide the specific decontamination design considerations as part of the SSC discussion (proposed 10 CFR 63.21(c)(18) [64 FR 8640]).

4.1.2.4 Non-Safety Structures, Systems, and Components

The information that must be provided for any non-safety SSCs discussed in this chapter includes:

Regulatory Bases–These are primarily the proposed 10 CFR 63 (64 FR 8640) requirements.

General Description–Provide a general description only to the extent that is sufficient to demonstrate the non-safety classification, which should be based on information contained in Section 1 of the related SDDs. Other diagrams may be included to show concepts or ideas to support the text to the extent needed to demonstrate the non-safety classification (proposed 10 CFR 63.112(a) [64 FR 8640]).

Decontamination

Information Required at the Time of Update to the License Application to Receive and Possess–Provide the specific decontamination design considerations as part of the SSC discussion (proposed 10 CFR 63.21(c)(18) [64 FR 8640]).

4.2 CARRIER PREPARATION BUILDING

Provide the required level of design detail for these systems, buildings, and equipment, as outlined in Section 4.1.1, Level of Design Detail, taking into consideration the following information.

Describe the purpose of the Carrier Preparation Building, including its role in the overall GROA operational process. Refer to the site map for the location of the building to show how the various waste handling systems work together and how they interface with other site systems. Identify and discuss general layout criteria that have been included in the design to ensure confinement of radioactivity. Discuss each of the systems in a sequential order. Identify the QL classification for each SSC discussed.

Consider the following outline for this section:

4.2.1 Carrier Preparation Building Materials Handling System
4.2.2 Electrical Systems
4.2.3 Fire Protection System
4.2.4 Non-Nuclear Heating, Ventilation, and Air-Conditioning System
4.2.5 Piped Utility Systems

4.3 CARRIER/CASK TRANSPORT SYSTEM

Provide the required level of design detail for these systems, buildings, and equipment, as outlined in Section 4.1.1, Level of Design Detail, taking into consideration the following information.

Describe the purpose of the Carrier/Cask Transport System, including its role in the overall GROA operational process. Refer to the site map for the location of the system, as appropriate, to show how the various waste handling systems work together and how they interface with other site systems. Identify and discuss general layout criteria that have been included in the design to ensure confinement of radioactivity. Identify the QL classification for each SSC discussed.

Consider the following outline for this section:

- 4.3.1 Carrier/Cask Rail System
- 4.3.2 Carrier/Cask Road System
- 4.3.3 On-Site Prime Mover System

4.4 WASTE HANDLING BUILDING

Provide the required level of design detail for these systems, buildings, and equipment, as outlined in Section 4.1.1, Level of Design Detail, taking into consideration the following information.

Describe the purpose of the waste handling building, including its role in the overall GROA operational process. Refer to the site map for the location of the buildings where these activities take place to show how the various waste handling systems work together and how they interface with other site systems. Identify and discuss general layout criteria that have been included in the design to ensure confinement of radioactivity. Discuss each system sequentially beginning with the receipt of the SNF or HLW and ending with their loading onto the WP transporter in preparation for emplacement underground. Clearly identify the QL classification for each SSC discussed.

Consider the following outline for this section:

- 4.4.1 Waste Handling Building Structure
- 4.4.2 Carrier/Cask Handling System
  - 4.4.2.1 Handling Equipment
  - 4.4.2.2 Supporting Equipment
- 4.4.3 Assembly Transfer System
  - 4.4.3.1 Assembly Drying System
  - 4.4.3.2 Assembly Handling System, Dry
  - 4.4.3.3 Assembly Handling System, Wet
  - 4.4.3.4 Basket Transport System
  - 4.4.3.5 Cask and Dual Purpose Canister Preparation System
  - 4.4.3.6 Cask Transport System

4.4.3.7 Control and Tracking System
4.4.3.8 Disposal Container Preparation System
4.4.3.9 Disposal Container Transport System
4.4.4 Canister Transfer System
4.4.4.1 Canister Handling System
4.4.4.2 Cask Preparation System
4.4.4.3 Control and Tracking System
4.4.4.4 Transport System
4.4.5 Disposal Container Handling System
4.4.5.1 Control and Tracking System
4.4.5.2 Disposal Container Handling System
4.4.5.3 Disposal Container Weld/Inspection System
4.4.5.4 Empty Disposal Container Preparation System
4.4.5.5 Waste Package Emplacement Preparation System
4.4.6 Waste Package Remediation System
4.4.6.1 Disposal Container/Waste Package Lid Removal System
4.4.6.2 Disposal Container/Waste Package Sampling System
4.4.6.3 Hot Cell Decontamination System
4.4.6.4 Hot Cell Hoist
4.4.6.5 Hot Cell Manipulator
4.4.7 Waste Handling Building Ventilation System
4.4.7.1 Confinement Area Ventilation System
4.4.7.2 Non-Confinement Area Ventilation System
4.4.8 Pool Water Treatment and Cooling System
4.4.8.1 Pool Water Cooling
4.4.8.2 Pool Water Leak Detection
4.4.8.3 Pool Water Level Control
4.4.8.4 Pool Water Makeup
4.4.9 Waste Handling Building Fire Protection System
4.4.9.1 Fire Detection System
4.4.9.2 Fire Suppression systems
4.4.10 Waste Handling Building Electrical System
4.4.10.1 Emergency Power Distribution
4.4.10.2 Emergency Power Source
4.4.10.3 Lightning Protection
4.4.10.4 Normal Power Distribution
4.4.10.5 Normal Power Source
4.4.11 Waste Handling Building Instrumentation and Control
4.4.11.1 Monitored Geological Repository Operations Monitoring and Control System
4.4.11.2 Performance Confirmation Data Acquisition/Monitoring System
4.4.12 Waste Handling Building Utility Systems
4.4.12.1 Facility Decontamination System
4.4.12.2 Piped Utility Systems
4.4.12.3 Process Supply Systems
4.4.12.4 Solid Sanitary Waste Collection System

4.5 WASTE TREATMENT BUILDING

State that this section provides a discussion of the Waste Treatment Building and associated SSCs. Explain that the SSCs involved in the solid, aqueous, and chemical low-level radioactive waste processing systems are discussed in Chapter 9 and will not be included in this section. Provide a reference to Chapter 9 for a discussion of the waste management systems and process and effluent monitoring and sampling systems. State that the source terms, design bases, parameters, system descriptions, capacities, and estimates of waste generation and processing are discussed in Chapter 9. State that dose assessments to the public are addressed in Chapter 7, and dose assessments to occupational workers are addressed in Chapter 10, which includes exposure from the waste management systems.

Provide the required level of design detail for this system as outlined in Section 4.1.1, Level of Design Detail, taking into consideration the following information.

Provide an overview of the waste management approach. Define the system scope including inputs, outputs, and interfaces with other systems such as utility and waste treatment, emergency and ventilation systems, and any other systems. Clearly identify the QL classification for each SSC discussed.

Consider the following outline for this section:

- 4.5.1 Waste Treatment Building
- 4.5.2 Radiological Monitoring System
- 4.5.3 Process Supply Systems
- 4.5.4 Piped Utility Systems
- 4.5.5 Fire Protection System
- 4.5.6 Electrical Systems
- 4.5.7 Waste Treatment Building Ventilation System
  - 4.5.7.1 Confinement Area Ventilation System
  - 4.5.7.2 Exhaust Stack Radiation
  - 4.5.7.3 Non-Confinement Area Ventilation System

4.6 BALANCE OF PLANT

Provide the required level of design detail for these systems, buildings, and equipment, as outlined in Section 4.1.1, Level of Design Detail, taking into consideration the following information.

State the purpose and refer to the site map for the location of the buildings or equipment discussed in this section. Discuss any related balance of plant activities that take place to show how they work together and interface with other site systems. Identify and discuss general layout criteria that have been included in the design to ensure confinement of radioactivity. Identify the QL classification for each SSC discussed.

Consider the following outline for this section:

4.6.1 Administration System
4.6.2 Emergency Response System
- 4.6.2.1 Medical System
- 4.6.2.2 Radiological Emergency Response System
- 4.6.2.3 Underground Response System

4.6.3 General Site Transportation System
4.6.4 Health Safety System
4.6.5 Monitored Geologic Repository Site Layout System
- 4.6.6.1 Site Drainage
- 4.6.6.2 Soil Stockpile

4.6.6 Maintenance and Supply System
4.6.7 Surface Environmental Monitoring System
- 4.6.7.1 Meteorological Monitoring System
- 4.6.7.2 Sample Collection System
- 4.6.7.3 Seismic Monitoring System

4.6.8 Site Electrical Power System
- 4.6.8.1 Grounding
- 4.6.8.2 Lighting
- 4.6.8.3 Lightning Protection
- 4.6.8.4 Power Distribution System
- 4.6.8.5 Power Transmission
- 4.6.8.6 Standby Power Source
- 4.6.8.7 Substations
- 4.6.8.8 Switchgear Building
- 4.6.8.9 Switchyard

4.6.9 Site Fire Protection System
4.6.10 Site Operations System
4.6.11 Site Radiological Monitoring System
- 4.6.11.1 Area Radiation Monitoring System
- 4.6.11.2 Continuous Air Monitoring System

4.6.12 Site Water System
4.6.13 Site Compressed Air System
- 4.6.13.1 Air Compression System
- 4.6.13.2 Industrial Air Distribution System
- 4.6.13.3 Instrument Air Distribution System

4.6.14 Site Communications
- 4.6.14.1 General Site Communications Systems
- 4.6.14.2 Microwave Systems

4.7 DECONTAMINATION OR DISMANTLEMENT OF SURFACE FACILITIES

Information Required at the Time of Construction Authorization–Describe the decommissioning goal (e.g., return to near greenfield condition). Describe in general the decommissioning process for the surface facilities. State that design features that facilitate decontamination of the surface facility SSCs include those that minimize the quantities of

radioactive wastes, contaminated equipment, secondary waste streams, and mixed hazardous and radioactive waste (proposed 10 CFR 63.21(c)(18) [64 FR 8640]).

Provide a general description of design considerations intended to facilitate decontamination or dismantlement of surface facilities. The following are examples of some design considerations that could be described to help to meet the goal of facilitating decommissioning (proposed 10 CFR 63.21(c)(18) [64 FR 8640]):

- Selection of materials and processes in surface facility construction and operations to minimize waste production
- Minimization of the absolute mass of shielding materials subject to neutron activation by placing the shielding as close to the SNF and HLW sources as practicable
- Use of modular design and inclusion of lifting points (to facilitate future demolition and removal of equipment and structures)
- Use of minimum surface roughness finishes on SSC surfaces that may be exposed to contamination
- Use of selected coatings to preclude penetration of radioactive gas, condensate, or deposited aerosols (if potentially present) into porous materials to permit future decontamination by a surface treatment
- Use of selected admixtures and mix design for concrete that may be exposed to contamination in order to reduce porosity
- Incorporation of features to collect and retain any possible leaks or spills.

Identify the design considerations that address the requirements of 10 CFR 20.1406 to design the facility for eventual decommissioning and that minimize, to the extent practicable, the generation of radioactive waste. The decommissioning criterion also is addressed in proposed 10 CFR 63.21(c)(18) and proposed 10 CFR 63.111(a) (64 FR 8640).

Information Required at the Time of Update to the License Application to Receive and Possess–Explain that the specific decontamination aspects of the affected SSCs are addressed in the sections where they are described.

4.8 REFERENCES

The following references were used to develop this chapter of the TGD. For the LA, this section will contain the references used to develop this chapter of the LA.

References Cited

CRWMS (Civilian Radioactive Waste Management System) M&O (Management and Operating Contractor) 1999. *Level of Design Detail Necessary for the License Application for Construction*

Authorization. B00000000-01717-1710-00003 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990708.0065.

DOE (U.S. Department of Energy) 1998. *Centralized Interim Storage Facility Topical Safety Analysis Report.* BA0000000-01717-5700-00017 REV 01. Two volumes. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.19990212.0117.

Dyer, J.R. and Horton, D.G. 1999. "Interim Guidance Pending Issuance of New Nuclear Regulatory Commission (NRC) Regulations for Yucca Mountain, Nevada." Letter from J.R. Dyer (DOE/YMSCO) to D.R. Wilkins (CRWMS M&O), June 18, 1999, OL&RC:AVG:1435, with enclosure, "Interim Guidance Pending Issuance of New NRC Regulations for Yucca Mountain." ACC: MOL.19990712.0039.

Codes, Standards, and Regulations

10 CFR (Code of Federal Regulations) 20. Energy: Standards for Protection Against Radiation. Readily available.

10 CFR 50. Energy: Domestic Licensing of Production and Utilization Facilities. Readily available.

64 FR (Federal Register) 8640. Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, Nevada. Proposed rule 10 CFR 63. Readily available.

INTENTIONALLY LEFT BLANK

CHAPTER 5. WASTE FORM AND WASTE PACKAGE DESIGN

Chapter 5 contains the description of the waste form and the waste package (WP) design as part of the engineered barrier system (EBS) (proposed 10 CFR 63.114(h) [64 FR 8640]). The EBS, except for the WP, is described in Chapter 6. Authors preparing the License Application (LA) shall use this document for guidance and must read the Introduction and Appendix B before writing their respective sections.

Proposed 10 CFR 63.24(b) (64 FR 8640) recognizes that all information may not be available at the time of docketing the LA. The guidance in this chapter adopts a differentiated approach for some required information. A differentiated approach allows only the information required at construction authorization (CA) versus that required at license to receive and possess high-level radioactive waste (HLW). Information in any section identified as "Information Required at the Time of Construction Authorization" will be provided at the time the LA is submitted for docketing. The information at the CA stage will be replaced or supplemented with additional information developed since docketing the LA. Some information required at the receive and possess stage may be totally new information. The additional information is identified as "Information Required at the Time of Update to the License Application to Receive and Possess." Information that is not differentiated is required at CA and must be updated as necessary at the time to receive and possess.

Because the WP design has no licensing precedent, the level of detail required in Chapter 5 is expected to be more than what is required for those structures, systems, and components (SSCs) that have licensing precedent. The focus of Chapter 5 must be to describe, with sufficient detail, the method for a representative WP design to demonstrate compliance.

5. PURPOSE AND SUMMARY

Provide an introductory discussion outlining the purpose and organization of this chapter. Provide a summary of the WP design and any conclusions regarding how the WP design method and specific WP designs support safe operation of the repository and the repository safety case.

Provide a summary of the role of the information contained in this chapter as it relates to the ability to safely operate the repository. Provide a summary of conclusions supporting the intent of this chapter and a description of important uncertainties that affect the conclusions. Explain how the WP design supports the overall U.S. Department of Energy (DOE) approach to repository safety strategy.

5.1 WASTE FORM

Provide a description of the radioactive waste and waste forms expected to be received, processed, and emplaced in the potential Yucca Mountain repository to demonstrate compliance with proposed 10 CFR 63.21(c)(12) (64 FR 8640). As defined by proposed 10 CFR 63.2 (64 FR 8640), "waste form" means the radioactive waste materials and any encapsulating or stabilizing matrix. This includes the commercial light water reactor spent nuclear fuel (SNF), DOE SNF, HLW from commercial and defense facilities, and Navy SNF (information on Navy SNF will be prepared in a classified addendum to the LA). Describe the design methods and

criteria for establishing representative groups of waste and waste forms that will encompass all wastes and waste forms to be placed in the repository. Explain how the criteria and analyses for these groups will envelop all wastes and waste forms, including those not identified at this time. Indicate that the DOE has considered all categories of SNF planned for disposal at the potential Yucca Mountain repository [CLST 3.4] (NRC 1998a). Also, indicate that the DOE has taken into account all types of HLW glass planned for Yucca Mountain disposal [CLST 4.1] (NRC 1998a). Describe in a tabulated form the kinds, sources, and amounts of waste and waste forms, including their physical, chemical, thermal, and radiological characteristics.

Describe any waste material testing program and the behavior of material in the expected repository environments. Indicate that the results of the test program are discussed as part of the total system performance assessment discussion in Chapter 8.

5.1.1 Commercial Spent Nuclear Fuel

Describe the amount and characteristics of commercial SNF for historical discharges and for forecast discharges. Information shall be presented as to the number of assemblies, burnup, and enrichment characteristics of the overall commercial waste stream. The method for discharge forecasts and forecast characteristics will not be covered in this section. Provide an overview of what is considered standard waste and how non-standard waste will be handled. The bounding characteristics of commercial SNF, in contrast to other waste types, shall be discussed.

5.1.2 U.S. Department of Energy Spent Nuclear Fuel

Describe the amount and characteristics of DOE SNF for historical discharges and for any forecast discharges. Provide information for all categories of waste as used for WP design [CLST 3.1] (NRC 1998a). The bounding characteristics of DOE SNF, in contrast to other waste types, shall be discussed.

5.1.3 High-Level Radioactive Waste

Discuss expected and general characteristics for HLW, including canister sizes and the number of canisters, to be received at the repository. Discuss how all types of HLW planned for disposal have been taken into account [CLST 4.1] (NRC 1998a). The bounding characteristics of HLW, in contrast to other waste types, shall be discussed.

5.1.4 Naval Waste

Discuss any unclassified information regarding the Navy fuel, including canister sizes and the number of canisters, expected to be received at the repository. The bounding characteristics of naval waste, in contrast to other waste types, shall be discussed.

5.1.5 Waste Form Testing Program

Describe the testing program for the waste forms to be received at the repository. Include discussions of tests for commercial SNF, DOE SNF, and HLW, and their relevance to performance under the wide range of conditions anticipated in the repository. Discussion shall include relevant information for illustrating that the DOE has conducted a consistent, sufficient,

and suitable testing program at the time of the LA submittal. State that the Navy, in a separate submittal to the U.S. Nuclear Regulatory Commission (NRC), will provide information on the naval fuel testing program.

Describe the results of the testing program and its relationship to the WP design, WP models, and performance assessment (PA). At a general level, provide an explanation of the test results supporting the models used to perform analyses. State that specific details are provided in Chapter 8. Analyses and models that have been used to predict future conditions and changes in the waste form should be supported by using an appropriate combination of such methods as field tests, in situ tests, laboratory tests that are representative of field conditions, monitoring data, and natural analog studies. State that the models used to predict the waste form degradation in the expected service environment are discussed in Chapter 8.

Discussion shall include justification for the use of SNF test results not specifically collected for Yucca Mountain for the environmental conditions expected to prevail at the site [CLST 3.5] (NRC 1998a), and for the use of test results for HLW not specifically collected for Yucca Mountain for environmental conditions expected to prevail after breaching of the containers at the site [CLST 4.5] (NRC 1998a). Include discussion to show how the DOE has conducted a consistent, sufficient, and suitable corrosion testing program for HLW and SNF [CLST 3.6] [CLST 4.6] (NRC 1998a).

Address the testing program that will continue after CA and the application of the data received from such tests.

5.2 WASTE PACKAGE DESIGN DESCRIPTION

This section describes the need for multiple WP designs and the suite of WP designs that are being considered to dispose of the various waste forms that will be received at the repository. Also, this section provides an overall description of the WP design and application of the design method to examples of representative WP designs. Specific aspects of application of the design method to each representative WP design will be included in Sections 5.3 through 5.8. This section will also include the design criteria and criterion bases common to all WP designs and design method.

State that in conformance with the postclosure performance objectives, the potential repository will include multiple barriers, including natural barriers and an EBS. State that the material in this chapter describes the design method and specific designs for the WP, including the key attributes of the WP design (proposed 10 CFR 63.113(a) [64 FR 8640]). Reference Section 2.2 of the LA for an overview of the DOE conformance with this regulation and the approach to the repository safety strategy. Also, reference Section 8.6 for additional information on the concept of multiple barriers and analyses of multiple barriers that demonstrate compliance with proposed 10 CFR 63.113(a) (64 FR 8640).

State that by the regulatory definition in proposed 10 CFR 63.2 (64 FR 8640), the WP is a part of the EBS. State that the design of the EBS outside the WP is discussed in Chapter 6 of the LA. Delineate the boundary between the WP and the remainder of the EBS discussed in Chapter 6.

Describe in general terms the relationships between the WP design, as described in this chapter, and the repository preclosure radiological safety assessment and postclosure PA, which are described in Chapters 7 and 8 of this document, respectively.

State that an explanation of measures used to support the models and abstractions, required by proposed 10 CFR 63.21(c)(9) (64 FR 8640), is provided in Chapter 8.

Describe the need for multiple WP designs for disposal of various waste forms. Describe in a tabular form multiple WP designs that the DOE is considering for the disposal of waste at the potential repository. Provide rationale for selection of the representative WP designs that will employ the design method to demonstrate compliance with the regulatory requirements at the time of CA. Describe the additional conceptual WP designs for unique waste forms that the DOE will consider for future disposal. Demonstrate that these additional designs will be similar to and have the same defense-in-depth features as the example designs described in Sections 5.5, 5.6, 5.7, and 5.8. Explain that the design for these additional WPs will be based on the method described in this chapter. State that detailed design description for these additional WPs will be included in the LA at the time to receive and possess or, as applicable, in future submittals under the proposed 10 CFR 63.44 (64 FR 8640) requirements.

The following general aspects of the WP design shall be addressed for the WP design data, method, models, and codes:

- State that collection, documentation, and development of all data, methods, models, or computer codes have been performed in accordance with the *Quality Assurance Requirements and Description* (DOE 1998) and associated implementing procedures [CLST General.1] (NRC 1998a), [RDTME 3.2.1] (NRC 1998b).

- State that expert elicitations are conducted and documented in accordance with the requirements of the *Quality Assurance Requirements and Description* (DOE 1998). Identify any expert elicitations that have been used in support of the WP design and the extent to which each was used [CLST General.2] (NRC 1998a), [RDTME 3.2.2] (NRC 1998b).

- Demonstrate that parameter values, assumed ranges, test data, probability distributions, and bounding assumptions used in the models are technically defensible and can reasonably account for known uncertainties [CLST General.5] (NRC 1998a).

Discuss analyses, tests, and test programs for the WP and its components. Such testing will include acceptance testing on the WP during and after construction of the WP, as well as testing during operation of the repository. Summarize the pertinent results of the analyses or tests performed to demonstrate WP confinement under normal preclosure conditions. Do not repeat the discussion of performance confirmation from Chapter 12 of the LA; rather, provide a reference to the appropriate section in Chapter 12, as applicable.

Briefly describe the design method and major components of the method, such as criticality, thermal, structural, and shielding considerations and material selection. Address the following:

- Discuss the aspects of the WP criticality control design. Reference Section 5.3.1 for a discussion of the WP design method for criticality control.
- Discuss the aspects of the WP thermal design. Reference Section 5.3.2 for a discussion of the WP structural design method.
- Discuss the aspects of the WP structural design. Reference Section 5.3.3 for a discussion of the WP structural design method.
- Discuss the aspects of the WP shielding design. Reference Section 5.3.4 for a discussion of the WP shielding design method.
- Discuss the aspects of the WP design related to WP materials. Reference Section 5.4 for a discussion of the WP material selection and performance.

Describe how the same method is applied to all WP designs and how it will be applied for designs still at the conceptual stage. Refer to Section 5.3 for the design method.

Summarize the major design features of each of the WP designs and their significant components. Include a general description of the WP characteristics, materials, and capacities. Include a brief description of how each significant component contributes to the overall WP design and to the acceptable performance of the WP. Provide a reference to Sections 5.5, 5.6, 5.7, and 5.8 in this chapter for additional information.

Briefly discuss operations related to the WP, including how WPs will be handled from receipt at the site, through loading with waste and sealing, to emplacement in a drift.

Briefly describe how the WP is designed to withstand natural phenomena, such as seismic events and operational accidents. Refer to Chapter 7 for the specific events considered and evaluated for design basis events (DBEs). Discuss the integrated safety analysis performed to ensure the preclosure performance objectives are satisfied. Briefly discuss how the integrated safety analysis results are reflected in this chapter. State that the WP design adequately addresses these DBEs, demonstrating that the performance objectives of the repository can be met, given the occurrence of these events.

Describe the evaluation of the compliance of the WP and its components with the design criteria using the design method described in Section 5.3. Also, discuss the interface between the design evaluation described in this section and the overall repository preclosure radiological safety analysis and postclosure PA described in Chapters 7 and 8, respectively.

Identify and justify selection of the variables, conditions, or other items that are determined to be probable subjects of technical specifications, particularly items that may influence the final WP design. Coordinate this discussion with the authors of Chapter 11, who will be tabulating similar information from all chapters of the LA.

5.2.1 Waste Package Design Overview

Provide the required level of design detail for the WP designs, as outlined in Section 5.2.3, taking into consideration the following information.

Provide a general description of the WP designs and the need for design variations to allow for disposal of numerous waste forms. Summarize the principal characteristics of four representative examples of WP designs that demonstrate implementation of the design analysis method.

Identify components that are important to radiological safety or waste isolation, indicate these items on drawings, and discuss their design and performance. Identify SSCs and system interfaces that are relied upon to maintain the capability of waste retrieval. Include a discussion of any coatings, liners, or fillers that may be incorporated in the WP design as it relates to safety. Discuss briefly the overall design features of the WP designs (e.g., absorber plates, thermal shunts, interlocking plates).

Provide the general criteria, criterion bases, and codes and standards common to the WP designs. State that specific WP design criteria and bases are described under specific WP design sections. Include, in summary tables, the gross weight; materials of construction; other materials, such as neutron absorbers; external dimensions and cavity size; internal structures; lifting devices; impact limiters, if applicable; amount of shielding; closures; means of confinement of radionuclides; and a description of how individual WPs will be uniquely identified.

5.2.2 General Design Criteria and Basis

Provide a list of the general design criteria that govern the WP designs provided in system description documents (SDDs). For each design criterion identified, include a discussion of the relationship of that criterion to the repository performance objectives. Provide a reference to Sections 5.4 through 5.8 for requirements applicable to the specific design considerations.

For preclosure WP design:

- Discuss the aspects of the WP design that will ensure there will not be uncontrolled release of radioactivity from the WP during the preclosure period. Include a discussion of protection against any postulated off-normal operations, WP internal changes, or external natural phenomena.

- Identify and discuss WP containment of radionuclides for normal conditions of storage. Identify the containment boundary of the WP and provide a summary of design specifications for these components. Identify penetrations in the primary containment boundary. Provide a summary of the performance specifications for components that penetrate the containment boundary. Identify all seals and welds that affect WP containment.

- Discuss the specific consideration given to interface control between WP design for handling, transport, and emplacement equipment, etc., that will prevent exceeding WP design bases and the breach of the WP in credible events.

For postclosure WP design:

- Describe the specific criteria and design consideration that will provide a high level of confidence that the WP designed and fabricated to those criteria will both meet the design criteria and the regulatory performance objectives in proposed 10 CFR 63.113(a) and proposed 10 CFR 63.113(b) (64 FR 8640).

- State that the results of design analyses will be used for total system performance assessment, as described in Chapter 8, that demonstrate compliance with this performance objective.

5.2.3 Level of Design Detail

This section provides the guidance for the required level of detail that must be provided for the WP. Note that there will not be a Section 5.2.3 on level of design detail in the LA.

Quality Level (QL) 1, 2, and 3 SSCs and non-safety SSCs are required to be described in the LA. See Section 2.4 of the TGD for more information related to QL classifications. For any SSC, the level of design detail required is dependent on its:

- Importance to protection of the public health and safety
- Importance to protection of worker health and safety
- Need to demonstrate compliance with regulatory requirements
- Need to support submittal of a docketable LA.

All major components of the WP have been identified as important to radiological safety and important to waste isolation; therefore, QL 1 applies to the WPs. The QL information that follows this paragraph is a listing of the specific information that must be included in the description of the WPs.

Quality Level 1: Waste Package

The following information must be provided for the WP. However, this information may be provided in various Chapter 5 sections, as applicable.

Regulatory Bases–These are primarily the proposed 10 CFR 63 (64 FR 8640) requirements.

Identify the WP as an SSC that is relied upon to limit or prevent potential accidents or mitigate their consequences. Refer to Chapter 7 for details on the analysis that identified the SSC as such. Describe the design features incorporated into the WP and describe the function of the WP, including controls that are relied upon to limit, prevent, or mitigate the consequences of DBEs. Use drawings as necessary to present information for the WP (Interim Guidance Section 112(e) [Dyer and Horton 1999]).

Discuss the WP design considerations until permanent closure that limit and control radiation exposures and radiation levels in restricted and unrestricted areas and releases of radioactive materials to unrestricted areas that could result in an annual dose of 0.25 mSv (25 mrem) to an individual member of the public at the boundary. Reference Chapter 10 for limits established by proposed 10 CFR 63.111(a) (64 FR 8640). Where appropriate, identify the sequence of events and how the WP responds to the event (proposed 10 CFR 63.111(b)(1) [64 FR 8640]).

Discuss the WP design considerations until permanent closure to limit the consequences of Category 2 DBEs. Identify the specific DBEs and reference Chapter 7 for the analysis of such events. Describe the design features and their functions incorporated into the WP that prevents or mitigates the consequences of the Category 2 DBEs (Interim Guidance Section 111(b)(2) [Dyer and Horton 1999]).

Describe the design criteria and design considerations that, when combined with the natural barriers, will demonstrate with reasonable assurance the ability of the potential repository to meet the postclosure performance objective specified in proposed 10 CFR 63.113(b) (64 FR 8640). This chapter must describe how the WP designs contribute to meeting the postclosure performance objective (proposed 10 CFR 63.113(b), proposed 10 CFR 63.21(c)(4)(i), and proposed 10 CFR 63.21(c)(4)(ii) [64 FR 8640]).

State that Chapter 8 will include a section assessing the degree to which features, events, and processes (FEPs) are expected to materially affect compliance with proposed 10 CFR 63.21(c)(5) (64 FR 8640), whether beneficial or potentially adverse to performance of the geologic repository. This will include technical basis for either inclusion or exclusion of specific FEPs of the geologic setting in the PA and inclusion or exclusion of degradation, deterioration, or alteration processes of the WP in the PA, including those processes that adversely would affect the performance of natural barriers (proposed 10 CFR 63.21(c)(5) [64 FR 8640]).

Provide a statement describing that Chapter 8 will provide the results of the PA of the potential geologic repository for the period after permanent closure, as required by proposed 10 CFR 63.113(c) (64 FR 8640) (Interim Guidance Section 21(c)(7) [Dyer and Horton 1999]).

Explain that Chapter 8 will describe the measures used to support the models and abstractions in compliance with proposed 10 CFR 63.114(g) (64 FR 8640) (proposed 10 CFR 63.21(c)(9) [64 FR 8640]).

Refer to Chapter 8 for a description of the use and incorporation of expert elicitations into assessments (proposed 10 CFR 63.21(c)(10) [64 FR 8640]).

Describe WP design considerations to permit implementation of a performance confirmation program. Describe the design features and their functions incorporated into the WP that permit implementation of a performance confirmation program (proposed 10 CFR 63.111(d) [64 FR 8640]).

License Specifications

- **Information Required at the Time of Construction Authorization**–When discussing equipment or parameters that are credited in the safety analysis for mitigating the consequences of a Category 1 or 2 DBE, note that the item is being considered as a subject for a license specification in accordance with proposed 10 CFR 63.21(c)(13) (64 FR 8640). Include the supporting information to demonstrate how and why it is credited in the safety analysis for mitigating the consequences of a Category 1 or 2 DBE. Reference Chapter 11 for an identification of the license specifications and a summary of their justification for being considered as a license specification (proposed 10 CFR 63.21(c)(13) [64 FR 8640]).

- **Information Required at the Time of Update to the License Application to Receive and Possess**–Update the information to reflect the latest status of WP related Category 1 DBEs (proposed 10 CFR 63.111(b)(1) [64 FR 8640]). Identify design parameters and features that are used to mitigate the consequences of a Category 1 or 2 DBE as being addressed in the license specifications, and reference Chapter 11 for a discussion of the license specification development process (proposed 10 CFR 63.21(c)(13) [64 FR 8640]).

Applicable Codes and Standards–List the codes and standards (including guidance documents and technical positions) required in the design and construction of the WP. The codes and standards should be listed on a structure or system level. The codes, standards, and other documents identified must be the same as those found in corresponding SDDs (proposed 10 CFR 63.21(c)(3) [64 FR 8640]).

Design Criteria–Identify and describe the principal design criteria for the WP designs from Section 1 of the relevant SDDs. For each principal design criterion identified, include a discussion of the relationship of that criterion to the performance objective specified in proposed 10 CFR 63.113(b) (proposed 10 CFR 63.21(c)(4)(i) [64 FR 8640]).

Design Bases–Identify and discuss the design bases and their relationship to the principal design criteria for the EBS. The project design bases are the combination of the system functions found in SDD Section 1.1 and the performance parameters found in SDD Sections 1.2.1 through 1.2.5 This definition is consistent with that of "design bases" found in proposed 10 CFR 63.2 (64 CFR 8640). The design criteria are also found in SDD Sections 1.2.1 through 1.2.5. The discussion should be based on the SDD criteria basis statements found in SDD Section 5 (proposed 10 CFR 63.21(c)(4)(ii) and proposed 10 CFR 63.114(j) [64 FR 8640]).

General Description–The general description shall be based on information contained in Section 1 of the related SDDs. The general system description shall provide a summary of the WP functions, the system design, concept of operations, and a description of system interfaces. This description should include a discussion of any special construction or fabrication techniques, unique testing programs, or special design and analysis procedures used for the WP, as applicable. Diagrams may be included to support the text (proposed 10 CFR 63.112(a), proposed 10 CFR 63.114(i), and proposed 10 CFR 63.114(j) [64 FR 8640]).

Provide a discussion of the materials of construction and the waste form (including components, general arrangement, and approximate dimensions) for the WP (proposed 10 CFR 63.21(c)(3) [64 FR 8640]).

Research and Development

- **Information Required at the Time of Construction Authorization**–Identify the required research and development (R&D) to confirm the adequacy of design. Provide available information that describes the type of R&D required and the reason the additional information is needed (proposed 10 CFR 63.21(c)(21) [64 FR 8640]). Refer to Chapter 11 for a description of the R&D program and the proposed schedule.

- **Information Required at the Time of Update to the License Application to Receive and Possess**–Provide the results from the required R&D identified in the LA at the time of CA. Discuss the reason the additional information was originally needed (proposed 10 CFR 63.21(c)(21) [64 FR 8640]).

Drawings and Diagrams–Simplified drawings, general arrangement drawings, and handling diagrams for the WP designs included in the LA are those that show information needed to support the safety case; other specified drawings are not required (proposed 10 CFR 63.112(a) [64 FR 8640]).

Retrieval–Address the following aspects of retrieval:

- **Information Required at the Time of Construction Authorization**–Describe the design features of the WPs that will permit or enable retrieval of any of the emplaced WPs during the specified retrieval period. The retrieval period starts when emplacement begins and continues for up to 50 years, unless the NRC establishes a different time period. Reference Section 11.1.6, as appropriate, for a discussion of the plan to accomplish waste retrieval (proposed 10 CFR 63.21(c)(19) and proposed 10 CFR 63.111(e)(1) [64 FR 8640]).

- **Information Required at the Time of Update to the License Application to Receive and Possess**–Update the retrieval discussion to reflect any additional information obtained during construction, including conformance of construction with the design; incorporate results of research programs conducted regarding retrieval; and incorporate any other relevant information regarding retrieval that was not available at the time of LA (CA) submittal. Cross-reference Chapter 11 as the location for the retrieval plan discussion (proposed 10 CFR 63.21(c)(19) and proposed 10 CFR 63.111(e)(1) [64 FR 8640]).

5.3 DESIGN METHOD

This section of the LA discusses the WP design method that will be used to design and develop WPs for the potential Yucca Mountain repository. The design method discussion must make a strong and convincing case that the method provides reasonable assurance that the proposed WP designs will support repository performance objectives and meet regulatory design requirements.

The focus of this section of the LA for CA will be to describe the design method used to demonstrate compliance of a set of four representative WP designs at the time of CA and any additional future WP designs. Also, this section must identify any additional information needed for the LA to receive and possess. The LA to receive and possess will be developed following the LA (CA) and during the repository construction. Sections 5.5 through 5.8 will provide a detailed description of the application of the design method to four representative examples of WP designs to demonstrate implementation of the design analysis method. Because the LA (CA) will present a limited, representative set of WP designs, the design method description must be of sufficient detail to enable acceptance by the NRC and for the use of the design method in future WP designs.

Provide the required level of design detail for the WP design method, as outlined in Section 5.2.3, taking into consideration information in the following paragraphs.

5.3.1 Criticality Design Method

Describe the criticality design method for the WP designs. The criticality design method description must make a strong and convincing case that the method provides reasonable assurance that the proposed WP designs will meet all preclosure and postclosure criticality design criteria.

Describe general criticality design criteria bases applicable to all WP designs. Specific criticality design criteria and bases for each WP design will be presented in Sections 5.5 through 5.8. Describe the purpose of the information in this section, as it relates to overall repository performance.

In describing the criticality design method and models, reference existing documents that detail the method approved by the NRC for WP criticality. Describe the process of using these documents in the analysis that shows the WP will meet the design criteria.

The following specific aspects of the criticality design should be included in this section:

- Describe conceptual models developed for evaluating criticality safety and determining quantitative values for the effective neutron multiplication factor (k_{eff}), including appropriate biases and uncertainties in the models [CLST 5.1] (NRC 1998a).

- Discuss evaluation of nuclear criticality occurring inside breached WPs by appropriate consideration of different types of SNF and vitrified waste, taking into account the design of the WP and other EBS components, and establishing adequate initial and boundary conditions for conceptual models and simulations [CLST 5.2] (NRC 1998a).

- Identify and describe credible scenarios that could result in criticality during postclosure, conservatively estimate their consequences, and identify and describe the means to mitigate unacceptable risks [CLST 5.3] (NRC 1998a).

- Identify and describe FEPs that may have an effect on nuclear criticality during the preclosure and postclosure regulatory periods and support such documentation with available data, analyses, and interpretations [CLST 5.4] (NRC 1998a).

5.3.2 Thermal Design Method

This section describes the thermal design method for the WP. The thermal design method description must make a strong and convincing case that the method provides reasonable assurance that the proposed WP designs will meet both preclosure and postclosure thermal design criteria.

Describe general thermal design criteria bases applicable to WP designs. Specific thermal design criteria bases for each WP design will be presented in Sections 5.5 through 5.8. Describe the purpose of the information in this section as it relates to overall repository performance.

Describe the methods and models used for evaluating the thermal design of the WP. Discuss the calculation method for effective thermal conductivity. Describe any software used for the thermal analysis of WPs. Include a statement regarding the qualification of the software. Describe the method of verification and validation of models used. Describe any significant assumptions used in the prediction of thermal values (e.g., projected waste form characteristics at repository arrival to determine waste heat-generation rates, the use of effective thermal conductivity, and the number of WPs in a drift representation).

5.3.3 Structural Design Method

This section describes the structural design method for the WP. The structural design method description must make a strong and convincing case that the method provides reasonable assurance that the proposed WP designs will withstand loads during normal handling operations and DBEs and meet the postclosure performance requirements.

Describe general structural design criteria bases applicable to all WP designs. Specific structural design criteria bases for each WP design will be presented in Sections 5.5 through 5.8. Include criteria that will be used for DBE evaluation. Describe the purpose of the information in this section as it relates to overall repository performance. As required by proposed 10 CFR 63.111(b)(1) (64 FR 8640), discuss the WP design considerations until permanent closure that will prevent radiation exposures and keep radiation levels from exceeding the limits in both restricted and unrestricted areas, and keep releases of radioactive materials to unrestricted areas from exceeding the limits specified in proposed 10 CFR 63.111(a) (64 FR 8640). For the structural criteria that require the WP to not be breached, give a detailed explanation of the method for determining that a breach would not occur.

Describe calculations performed for relatively simple structural load conditions, for which a formula has been developed. Describe the finite-element program software used for complex structural analyses of the WP designs. Show that the models have been validated and the program software have been appropriately qualified under approved quality assurance and control procedures.

Describe (in coordination with Chapters 4 and 6 authors) the handling operations expected during normal repository operations. Describe the evaluations used to determine the limiting loads expected during repository operations. Describe the analyses performed to provide reasonable assurance that the WP meets the design requirements.

Describe the DBE selection process and supporting analyses for the WP designs. Describe the screening process and results for external and internal DBEs as related to WP designs. Describe the process and results of bounding DBE identification. For each DBE, describe the course of the event, and describe the analysis to be performed to provide reasonable assurance that the WP meets the design requirements.

This section must address the following specific aspects of the structural design concerning the effects of seismically induced rockfall on WP performance:

- Show that the seismic hazard inputs used to estimate rockfall potential are consistent with the inputs used in the design and PAs as established in the DOE topical report addressing design and PA inputs [RDTME 3.2.3] (NRC 1998b).

- State that the size distribution of rocks that may potentially fall on the WPs is estimated from site-specific data (e.g., distribution of joint patterns, spacing, orientation in three dimensions) with adequate consideration of associated uncertainties [RDTME 3.2.4] (NRC 1998b). Alternatively, refer to a Chapter 8 section for this information if this discussion is provided in that chapter.

- Show that the analytical model used in the estimation of impact load due to rockfall on the WP is based on reasonable assumptions and site data, consistent with the emplacement drift and WP designs, and defensible with respect to providing realistic or bounding estimates or impact loads and stresses [RDTME 3.2.5] (NRC 1998b). Alternatively, refer to a Chapter 8 section for this information if this discussion is provided in that chapter.

- Discuss how the thermal-mechanical analyses that provide the background conditions upon which seismic loads are superimposed consider time-dependent jointed rock behavior [RDTME 3.2.6] (NRC 1998b).

- Show that rockfall analyses consider, in a rational and realistic way through dynamic analyses, the possibility of multiple blocks falling onto a WP simultaneously, and the extent of potential rockfall area around an individual emplacement drift, as well as over the entire repository as functions of ground motions [RDTME 3.2.7] (NRC 1998b). Alternatively, refer to a Chapter 8 section for this information if this discussion is provided in that chapter.

5.3.4 Shielding and Source Term Design Method

This section describes the shielding and source term design method for the WP designs. This design method description must make a strong and convincing case that the method provides

reasonable assurance that the proposed WP designs will meet both preclosure and postclosure shielding design criteria.

Describe general shielding design criteria bases applicable to the WP designs. Specific shielding design criteria and bases for each WP design will be presented in Sections 5.5 through 5.8. Describe the purpose of the information in this section as it relates to overall repository performance. Describe how shielding and source term contribute to other aspects of the repository designs, specifically with respect to personnel and equipment radiation and radiolysis-enhanced corrosion. Discuss the importance of shielding calculations for both preclosure and postclosure repository performances.

Discuss applicable industry codes and standards that were used in the shielding calculations. Identify the value of design parameters used to meet the WP design criteria, and describe any uncertainties associated with the parameters and the treatment of those uncertainties.

Describe the method and model used in the shielding calculations. Include appropriate sketches and dimensions of the radial and axial shielding materials. Identify differences in the models for normal and accident conditions. Reference sources of data, and include a statement regarding the qualification of data. Input and output data and interpretations should also be provided, along with the basis for the interpretation. Document the role of expert elicitation, if it was used.

Provide a general description of the basic method and models used to determine the gamma and neutron dose rates at selected points outside the WP. Discuss basic input parameters in detail. Include a description of the spatial source distribution and any computer program used. Describe and reference the program software validation and verification documentation. Provide the basis for selecting the program, attenuation and removal cross sections, and buildup factors. Tabulate flux-to-dose rate conversion factors as a function of energy, and provide appropriate references.

Discuss the source term calculation method. List software used, assumptions made, and uncertainties involved, including a discussion of validation and qualification of software, assumptions, and data. Evaluate the source term model with respect to the WP functions and criteria. Describe and document the design evaluations that the source term model feeds.

5.4 WASTE PACKAGE MATERIAL SELECTION AND PERFORMANCE

Discuss specific aspects of the WP designs related to WP materials. Briefly discuss the material selection features of the WP. Indicate that the discussion of the fabrication of the WP and the closure weld is also provided in this section.

Provide the required level of design detail for the WP material selection and performance, as outlined in Section 5.2.3, taking into consideration the following information.

5.4.1 Design Basis

This section provides the design basis for the selection of the WP materials. Discussion should focus on WP materials that support waste isolation. WP components that function primarily for

heat transfer, criticality control, etc., may be discussed at a general level, but specific detail is to be provided in the respective WP design Sections 5.5 through 5.8.

Discuss the relationship between the WP material selection and the performance objectives specified in proposed 10 CFR 63.111(b)(1), proposed 10 CFR 63.113(a), proposed 10 CFR 63.113(b) (64 FR 8640); and Interim Guidance Section 111(b)(2) (Dyer and Horton 1999). Provide sufficient detail on the behavior of the WP materials and their compatibility in the expected service environment during normal operations and post-accident situations. Include the anticipated environment and degradation modes for each WP component. Identify and describe the broad range of environmental conditions within the WP emplacement drifts that may promote the corrosion processes, taking into account possible irregular wet and dry cycles that may enhance the rate of container degradation [CLST 1.2] (NRC 1998a). Demonstrate that the DOE has identified and considered likely modes of corrosion for container materials, including dry-air oxidation, humid-air corrosion, and aqueous corrosion processes such as general corrosion, localized corrosion, microbial-induced corrosion, stress corrosion cracking, hydrogen embrittlement, and the effect of galvanic coupling [CLST 1.1] (NRC 1998a).

Discuss the design basis and applicable codes and standards for disposal container materials based on information from the SDDs. Verify conformance of the materials of the disposal container and its components with the design criteria, as outlined in the SDDs.

Discuss the design basis, codes, and standards for the disposal container fabrication, including the closure weld, based on information from the SDDs.

5.4.2 Design Description

Describe the materials for the significant WP components. Include tables with material properties and allowable stresses and strains associated with temperature, as appropriate. Discuss and report in this section appropriate corrosion allowances, and show how they were used in structural analyses in Section 5.3.3 of the LA.

Describe the material selection process used for WP components and alternative materials considered. Show a defensible rationale, including safety considerations and the practicality of the chosen design materials.

Discuss short-term and long-term material tests and their relevance to the selection of the materials and their performance under the wide range of conditions anticipated in the repository. Discussion shall include relevant information concerning the identification and consideration of relevant mechanical failure processes that may affect the performance of the proposed container materials [CLST 2.1] (NRC 1998a). Discuss the identification and consideration of the effect of material stability on mechanical failure processes for various container materials as a result of prolonged exposure to the expected range of temperatures and stresses, including the effects of chemical composition, microstructure, thermal treatments, and fabrication processes [CLST 2.2] (NRC 1998a).

Address the need for the material testing program that continues after CA and the application of the data received from such tests. Discuss relevant information for illustrating that the DOE has

conducted a consistent, sufficient, and suitable corrosion testing program at the time of LA submittal. In addition, discuss specific plans for further testing to reduce any significant areas of uncertainty as part of the performance confirmation program [CLST 1.6] [CLST 2.6] [CLST 3.6] [CLST 4.6] [CLST 6.6] (NRC 1998a).

Determine if such tests will be used for performance confirmation and refer to Chapter 12 for additional information. Include relevant information for demonstrating that the DOE has established a defensible program of corrosion monitoring and testing of the engineered subsystems components during the performance confirmation period to ensure they are functioning as intended and anticipated [CLST 1.7] (NRC 1998a). In addition, discuss information to demonstrate that the DOE has established a defensible program of monitoring and mechanical testing of the engineered subsystem components during the performance confirmation period to ensure they are functioning as intended and anticipated in the presence of thermal and stress perturbations [CLST 2.7] (NRC 1998a).

Identify the specifications of the materials and general manufacturing methods to be used, including the process for joining the inner and outer barriers of the disposal container and the welding process to be used. Discuss how these processes have been qualified. Include reference to or a discussion of methods and procedures used for compliance with the applicable codes and standards. Discussion shall include relevant information for demonstrating that the DOE has considered the compatibility of container materials and the variability in container manufacturing processes, including welding, in WP failure analyses and in the evaluation of radionuclide release [CLST 2.4] [CLST 3.4] (NRC 1998a).

Discuss analyses or tests for the WP and its components. Such testing will include acceptance testing on the WP during and after fabrication of the WP, as well as testing during operation of the repository. Do not repeat the discussion of performance confirmation from Chapter 12 of the LA; rather, provide a reference to the appropriate section in Chapter 12, as applicable.

Discuss WP testing acceptance criteria and the WP maintenance program. Discuss the basis for the acceptance criteria. Discuss analyses or tests to be performed prior to the first use of the WP to show that the acceptance criteria are met. Discuss visual inspections to be performed and the intended purpose for each. Provide acceptance criteria for each of these inspections, as well as the action to be taken if noncompliance is encountered.

Describe the leak tests to be performed. Describe the acceptance criteria and the action to be taken if the criteria are not met. Identify structures and components of the WP designs that require further R&D to confirm design adequacy. This discussion complies with proposed 10 CFR 63.21(c)(21) (64 FR 8640). Coordinate this discussion with the authors of Chapter 11, who will be summarizing the information from all chapters of the LA.

Describe protection of unstabilized stainless steels or other materials from sensitization if such materials are used in disposal container construction.

5.4.3 Design Evaluation

Describe the evaluation of the compliance of the materials of the WP and its components with the approved design criteria. Describe the interface between the design evaluation described in this section and the overall repository preclosure radiological safety analysis and postclosure PA described in Chapters 7 and 8. Reference Section 8.3 for a qualitative description of the expected performance of materials in the long-term repository environment.

At a general level, for both design applications and accident analyses, provide an explanation of measures supporting the models used to perform analyses. Analyses and models that have been used to predict future conditions and changes in the system should be supported by using an appropriate combination of such methods as field, in situ, and laboratory tests that are representative of field conditions, monitoring data, and natural analog studies. Discussion shall include justification for the use of data collected in corrosion tests not specifically designed or performed for the Yucca Mountain repository program for environmental conditions expected to prevail at the Yucca Mountain site [CLST 1.5] [CLST 2.5] (NRC 1998a).

Discuss possible chemical, galvanic, or other reactions, such as gamma-induced radiolysis, in the WP. For each component material in the WP, list all chemically or galvanically dissimilar materials with which the component has contact. Describe any measures to be taken to prevent contact or reaction between materials, and discuss the predicted or known effectiveness of such measures.

Demonstrate that the numerical models used for container materials stability and mechanical failures are effective representations, taking into consideration associated uncertainties, of the expected materials behavior and are not likely to underestimate the actual rate of failure in the repository environment [CLST 2.3] (NRC 1998a).

Demonstrate that the DOE has considered the compatibility of container materials, the range of material conditions, and the variability in container fabrication processes, including welding, in assessing the performance expected in the container's intended waste isolation function [CLST 1.4] (NRC 1998a).

Demonstrate that WP materials are compatible with environments to be experienced during loading operations and after emplacement. Discuss potential reactions in the presence of liquids used in loading; include discussion of chemical and galvanic actions, production of explosive or toxic gases, and degradation.

Discuss that the DOE has considered the compatibility of SNF and the internal components of the WP, such as the basket materials in the evaluation of radionuclide releases. Specifically, demonstrate that the SNF shall not compromise the performance of the WP [CLST 3.4] (NRC 1998a); that the DOE has assessed the compatibility of HLW glass with internal components of the WP in the evaluation of radionuclide release, taking into account codisposal with SNF owned by the DOE in the same WP; and that HLW glass shall not compromise the performance of the WP [CLST 4.4] (NRC 1998a). Also, provide a discussion to justify the use of test results for HLW glass not specifically collected for the Yucca Mountain site for

environmental conditions expected to prevail after breaching of the containers at the Yucca Mountain site [CLST 4.5] (NRC 1998a).

As relevant to the WP design, include discussion on the following to ensure that:

- The DOE has identified and considered the effects of drip shield and backfill and the timing of their emplacement on the thermal loading of the repository, WP lifetime (including container corrosion and mechanical failure), and the release of radionuclides from the EBS [CLST 6.1] (NRC 1998a). Reference Section 6.3 for detailed information on drip shield and backfill.

- If ceramic coating is considered for WP design, the DOE has identified and considered the effects of ceramic coating on WP lifetime, including negative consequences as a result of breakdown of the ceramic coating (cracking, spalling, or delamination) in response to the action of environment, manufacturing defects, mechanical impacts, stresses arising from a multiplicity of sources, and the potential for enhanced localized corrosion of the containers that might occur at cracks or perforations in the ceramic coating [CLST 6.2] (NRC 1998a).

- The DOE has evaluated the compatibility of ceramic coating materials, if they are considered for WP design, with outer overpack materials and the combined effect of ceramic coating with backfill on container lifetime [CLST 6.3] (NRC 1998a).

- The DOE has identified and considered the effects of drip shields (with backfill) on WP lifetime, including extension of the humid-air corrosion regime, environmental effects, breakdown of drip shields and resulting mechanical impacts on the WP, the potential for crevice corrosion at the junction between the WP and the drip shield, and the potential for condensate formation and dripping on the underside of the shield [CLST 6.4] (NRC 1998a).

- The DOE has justified the use of test results for drip shields, ceramic coatings, and backfill materials not specifically collected for the Yucca Mountain site for the environmental conditions expected to prevail at the potential Yucca Mountain repository [CLST 6.5] (NRC 1998a).

5.5 WASTE PACKAGE DESIGN FOR 21 PRESSURIZED-WATER REACTOR SPENT NUCLEAR FUEL ASSEMBLIES

Provide the required level of design detail for the 21 pressurized-water reactor SNF assemblies WP design, as outlined in Section 5.2.3, taking into consideration the following information.

Provide a description of the design features for the WP and its components and, for the 21 pressurized-water reactor SNF assemblies WP, include specific design information, design criteria and bases, and design and performance issues.

Include a description of the WP characteristics and materials. Reference Section 5.1 for a general description of the 21 pressurized-water reactor SNF. Discuss proposed loading limits for

the 21 pressurized-water reactor SNF assemblies WP based on thermal and criticality evaluations. In the description, state the functions that each WP component is to perform and describe the range of environmental conditions needed by the WP to perform its functions.

Include the WP capacity, gross weight, materials of construction, other materials (e.g., neutron absorbers), external dimensions, cavity size, internal structures, any openings, means of passive heat dissipation, outer and inner protrusions, lifting devices, impact limiters (if applicable), amount of shielding, closures, means of confinement of radionuclides, and a description of how individual WPs will be uniquely identified. Identify any components that are important to radiological safety or waste isolation, indicate these items on drawings, and discuss their design and performance. Include a discussion of any coatings, liners, or fillers that may be incorporated in the WP design. Identify the design parameter values used to meet the design criteria. Describe any uncertainties associated with the parameters and the treatment of those uncertainties. Discuss the aspects of the design that will ensure there will not be uncontrolled release of radioactivity from the WP. Discuss specific WP performance for identified bounding DBEs or internal changes.

Discuss specific consideration given to interface control between WP design and design for handling, transport, and emplacement equipment that will prevent exceeding WP design bases and breaching of WPs during credible events.

For preclosure and postclosure design, describe the specific criteria and design criteria considerations that will provide a high level of confidence that WPs designed and fabricated to those criteria will both meet the design criteria and the regulatory performance objectives in proposed 10 CFR 63.111(b)(1), proposed 10 CFR 63.111(d), proposed 10 CFR 63.111(e)(1), proposed 10 CFR 63.113(a), proposed 10 CFR 63.113(b) (64 FR 8640); and Interim Guidance Section 111(b)(2) (Dyer and Horton 1999).

Describe the evaluation of the compliance of the WP and its components with the design criteria using the design methodology described in Section 5.3. Also, discuss the interface between the design evaluation described in this section and the overall repository preclosure radiological safety analysis and postclosure PA described in Chapters 7 and 8, respectively.

Discuss specific analyses and tests for the WP and its components that are not included in Sections 5.2, 5.3, and 5.4. Such testing will include acceptance testing during and after construction of the WP, as well as testing during operation of the repository. Discuss specific design measures such as instrumentation intended for performance confirmation. Do not repeat the discussion of performance confirmation in Chapter 12 of the LA; rather, provide a reference to the appropriate section in Chapter 12, as applicable.

Describe modeling methods used to demonstrate that the design parameters are met, as required by proposed 10 CFR 63.21(c)(9) (64 FR 8640). State that measures used to support the methods and abstractions are described in Chapter 8.

Discuss the specific aspects of the WP design that are related to WP materials. Reference Section 5.4 for a discussion of the WP material selection and performance.

Discuss the specific aspects of the WP criticality control design. Discuss the aspects of the design that will ensure that the design will meet the defined criticality control requirements for preclosure and postclosure operations.

Discuss the specific aspects of the WP structural design. Identify and discuss WP containment of pressurized-water reactor fuel for normal operations. Identify the structural components of the WP and provide a summary of design specifications for these components. Identify seals and welds that affect WP containment.

Discuss the specific aspects of the WP shielding design. Describe the method and results for source term development. Discuss the expected dose rates for the WP.

5.6 WASTE PACKAGE DESIGN FOR 44 BOILING-WATER REACTOR SPENT NUCLEAR FUEL ASSEMBLIES

Provide the required level of design detail for the 44 boiling-water reactor SNF assemblies WP design, as outlined in Section 5.2.3, taking into consideration the following information.

Provide a description of the design features for the WP and components and, for the 44 boiling-water reactor SNF assemblies WP, include specific design information, design criteria and bases, and design and performance issues.

Include a description of the WP characteristics and materials. Reference Section 5.1 for a general description of the 44 boiling-water reactor SNF. Discuss proposed loading limits for the 44 boiling-water reactor SNF assemblies WP based on thermal and criticality evaluations. In the description, state the functions that each WP component is to perform and describe the range of environmental conditions needed by the WP to perform its functions.

Include the WP capacity, gross weight, materials of construction, other materials (e.g., neutron absorbers), external dimensions, cavity size, internal structures, any openings, means of passive heat dissipation, outer and inner protrusions, lifting devices, impact limiters (if applicable), amount of shielding, closures, means of confinement of radionuclides, and a description of how individual WPs will be uniquely identified. Identify any components that are important to radiological safety or waste isolation, indicate these items on drawings, and discuss their design and performance. Include a discussion of any coatings, liners, or fillers that may be incorporated in the WP design. Identify the design parameter values used to meet the design criteria. Describe any uncertainties associated with the parameters and the treatment of those uncertainties.

Discuss the aspects of the design that will ensure there will not be uncontrolled release of radioactivity from the WP. Discuss specific WP performance for identified bounding DBEs or internal changes.

Discuss specific consideration given to interface control between WP design and design for handling, transport, and emplacement equipment that will prevent exceeding WP design bases and breaching of WPs during credible events.

For preclosure and postclosure design, describe the specific criteria and design criteria considerations that will provide a high level of confidence that WPs designed and fabricated to those criteria will both meet the design criteria and the regulatory performance objectives in proposed 10 CFR 63.111(b)(1), proposed 10 CFR 63.111(d), proposed 10 CFR 63.111(e)(1), proposed 10 CFR 63.113(a), proposed 10 CFR 63.113(b) (64 FR 8640); and Interim Guidance Section 111(b)(2) (Dyer and Horton 1999).

Describe the evaluation of the compliance of the WP and its components with the design criteria using the design methodology described in Section 5.3. Also, discuss the interface between the design evaluation described in this section and the overall repository preclosure radiological safety analysis and postclosure PA described in Chapters 7 and 8, respectively.

Discuss specific analyses and tests for the WP and its components that are not included in Sections 5.2, 5.3, and 5.4. Such testing will include acceptance testing during and after construction of the WP, as well as testing during operation of the repository. Discuss specific design measures such as instrumentation intended for performance confirmation. Do not repeat the discussion of performance confirmation in Chapter 12 of the LA; rather, provide a reference to the appropriate section in Chapter 12, as applicable.

Describe modeling methods used to demonstrate that the design parameters are met as required by proposed 10 CFR 63.21(c)(9) (64 FR 8640). State that measures used to support the methods and abstractions are described in Chapter 8.

Discuss the specific aspects of the WP design that are related to WP materials. Reference Section 5.4 for a discussion of the WP material selection and performance.

Discuss the specific aspects of the WP criticality control design. Discuss the aspects of the design that will ensure that the design will meet the defined criticality control requirements for preclosure and postclosure operations.

Discuss the specific aspects of the WP structural design. Identify and discuss WP containment of boiling-water reactor fuel for normal operations. Identify the structural components of the WP, and provide a summary of design specifications for these components. Identify seals and welds that affect WP containment.

Discuss the specific aspects of the WP shielding design. Describe the method and results for source term development. Discuss the expected dose rates for the WP.

5.7 WASTE PACKAGE DESIGN FOR U.S. DEPARTMENT OF ENERGY DEFENSE HIGH-LEVEL RADIOACTIVE WASTE AND SPENT NUCLEAR FUEL

Provide the required level of design detail for the DOE defense HLW and SNF WP design, as outlined in Section 5.2.3, taking into consideration the following information.

Provide a description of the design features for the WP and components for the DOE HLW and SNF co-disposal WP. Include specific design information, design criteria and bases, and design and performance issues.

Include a description of the WP characteristics and materials. Reference Section 5.1 for a general description of DOE HLW and SNF. Discuss proposed loading limits for the DOE HLW and SNF WP based on thermal and criticality evaluations. In the description, state the functions that each WP component is to perform and describe the range of environmental conditions needed by the WP to perform its functions.

Include the WP capacity, gross weight, materials of construction, other materials (e.g., neutron absorbers), external dimensions, cavity size, internal structures, any openings, means of passive heat dissipation, outer and inner protrusions, lifting devices, impact limiters (if applicable), amount of shielding, closures, means of confinement of radionuclides, and a description of how individual WPs will be uniquely identified. Identify any components that are important to radiological safety or waste isolation, indicate these items on drawings, and discuss their design and performance. Include a discussion of any coatings, liners, or fillers that may be incorporated in the WP design. Identify the design parameter values used to meet the design criteria. Describe any uncertainties associated with the parameters and the treatment of those uncertainties.

Discuss the aspects of the design that will ensure there will not be uncontrolled release of radioactivity from the WP. Discuss specific WP performance for identified bounding DBEs or internal changes.

Discuss specific consideration given to interface control between WP design and design for handling, transport, and emplacement equipment that will prevent exceeding WP design bases and breaching of WPs during credible events.

For preclosure and postclosure design, describe the specific criteria and design criteria considerations that will provide a high level of confidence that WPs designed and fabricated to those criteria will both meet the design criteria and the regulatory performance objectives in proposed 10 CFR 63.111(b)(1), proposed 10 CFR 63.111(d), proposed 10 CFR 63.111(e)(1), proposed 10 CFR 63.113(a), proposed 10 CFR 63.113(b) (64 FR 8640); and Interim Guidance Section 111(b)(2) (Dyer and Horton 1999).

Describe the evaluation of the compliance of the WP and its components with the design criteria using the design methodology described in Section 5.3. Also, discuss the interface between the design evaluation described in this section and the overall repository preclosure radiological safety analysis and postclosure PA described in Chapters 7 and 8, respectively.

Discuss specific analyses and tests for the WP and its components that are not included in Sections 5.2, 5.3, and 5.4. Such testing will include acceptance testing during and after construction of the WP, as well as testing during operation of the repository. Discuss specific design measures such as instrumentation intended for performance confirmation. Do not repeat the discussion of performance confirmation in Chapter 12 of the LA; rather, provide a reference to the appropriate section in Chapter 12, as applicable.

Describe modeling methods used to demonstrate that the design parameters are met as required by proposed 10 CFR 63.21(c)(9) (64 FR 8640). State that measures used to support the methods and abstractions are described in Chapter 8.

Discuss the specific aspects of the WP design that are related to WP materials. Reference Section 5.4 for a discussion of the WP material selection and performance.

Discuss the specific aspects of the WP criticality control design. Discuss the aspects of the design that will ensure that the design will meet the defined criticality control requirements for preclosure and postclosure operations.

Discuss the specific aspects of the WP structural design. Identify and discuss WP containment of DOE HLW and SNF for normal operations. Identify the structural components of the WP, and provide a summary of design specifications for these components. Identify seals and welds that affect WP containment.

Discuss the specific aspects of the WP shielding design. Describe the method and results for source term development. Discuss the expected dose rates for the WP.

5.8 WASTE PACKAGE DESIGN FOR NAVAL FUEL

Naval fuel will arrive at the potential repository in canisters suitable for long-term disposal. The Navy, in a separate submittal, will provide classified information on the naval fuel material to the NRC. Provide the required level of unclassified design detail for this WP design, as outlined in Section 5.2.3, taking into consideration the following information.

Provide a description of the design features for the WP and components for the naval fuel WP. Include specific design information, design criteria and bases, and design and performance issues.

Include a description of the WP characteristics and materials. Reference Section 5.1 for a general description of naval fuel. Discuss proposed loading limits for naval fuel based on thermal and criticality evaluations. In the description, state the functions that each WP component is to perform and describe the range of environmental conditions needed by the WP to perform its functions.

Include the WP capacity, gross weight, materials of construction, other materials (e.g., neutron absorbers), external dimensions, cavity size, internal structures, any openings, means of passive heat dissipation, outer and inner protrusions, lifting devices, impact limiters (if applicable), amount of shielding, closures, means of confinement of radionuclides, and a description of how individual WPs will be uniquely identified. Identify any components that are important to radiological safety or waste isolation, indicate these items on drawings, and discuss their design and performance. Include a discussion of any coatings, liners, or fillers that may be incorporated in the WP design. Identify the design parameter values used to meet the design criteria. Describe any uncertainties associated with the parameters and the treatment of those uncertainties.

Discuss the aspects of the design that will ensure there will not be uncontrolled release of radioactivity from the WP. Discuss specific WP performance for identified bounding DBEs or internal changes.

Discuss specific consideration given to interface control between WP design and design for handling, transport, and emplacement equipment that will prevent exceeding WP design bases and breaching of WPs during credible events.

For preclosure and postclosure design, describe the specific criteria and design criteria considerations that will provide a high level of confidence that WPs designed and fabricated to those criteria will both meet the design criteria and the regulatory performance objectives in proposed 10 CFR 63.111(b)(1), proposed 10 CFR 63.111(d), proposed 10 CFR 63.111(e)(1), proposed 10 CFR 63.113(a), proposed 10 CFR 63.113(b) (64 FR 8640); and Interim Guidance Section 111(b)(2) (Dyer and Horton 1999).

Describe the evaluation of the compliance of the WP and its components with the design criteria using the design methodology described in Section 5.3. Also, discuss the interface between the design evaluation described in this section and the overall repository preclosure radiological safety analysis and postclosure PA described in Chapters 7 and 8, respectively.

Discuss specific analyses and tests for the WP and its components that are not included in Sections 5.2, 5.3, and 5.4. Such testing will include acceptance testing during and after construction of the WP, as well as testing during operation of the repository. Discuss specific design measures such as instrumentation intended for performance confirmation. Do not repeat the discussion of performance confirmation in Chapter 12 of the LA; rather, provide a reference to the appropriate section in Chapter 12, as applicable.

Describe modeling methods used to demonstrate that the design parameters are met as required by proposed 10 CFR 63.21(c)(9) (64 FR 8640). State that measures used to support the methods and abstractions are described in Chapter 8.

Discuss the specific aspects of the WP design that are related to WP materials. Reference Section 5.4 for a discussion of the WP material selection and performance.

Discuss the specific aspects of the WP criticality control design. Discuss the aspects of the design that will ensure that the design will meet the defined criticality control requirements for preclosure and postclosure operations.

Discuss the specific aspects of the WP structural design. Identify and discuss WP containment of naval fuel for normal operations. Identify the structural components of the WP, and provide a summary of design specifications for these components. Identify seals and welds that affect WP containment.

Discuss the specific aspects of the WP shielding design. Describe the method and results for source term development. Discuss the expected dose rates for the WP.

5.9 REFERENCES

The following references were used to develop this chapter of the TGD. For the LA, this section will contain the references used to develop this chapter of the LA.

References Cited

DOE (U.S. Department of Energy) 1998. *Quality Assurance Requirements and Description.* DOE/RW-0333P, Rev. 8. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.19980601.0022.

Dyer, J.R. and Horton, D.G. 1999. "Interim Guidance Pending Issuance of New Nuclear Regulatory Commission (NRC) Regulations for Yucca Mountain, Nevada." Letter from J.R. Dyer (DOE/YMSCO) to D.R. Wilkins (CRWMS M&O), June 18, 1999, OL&RC:AVG:1435, with enclosure, "Interim Guidance Pending Issuance of New NRC Regulations for Yucca Mountain." ACC: MOL.19990712.0039.

NRC (U.S. Nuclear Regulatory Commission) 1998a. *Issue Resolution Status Report Key Technical Issue: Container Life and Source Term.* Rev. 0. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.19980416.0784.

NRC 1998b. *Issue Resolution Status Report Key Technical Issue: Repository Design and Thermal-Mechanical Effects.* Rev. 1. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.19981130.0219.

Codes, Standards, and Regulations

64 FR (Federal Register) 8640. Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, Nevada. Proposed rule 10 CFR 63. Readily available.

INTENTIONALLY LEFT BLANK

CHAPTER 6. ENGINEERED BARRIER SYSTEM DESIGN (EXCLUDING WASTE PACKAGE)

This chapter provides guidance for the authors of Chapter 6 of the License Application (LA) to describe the engineered barrier system (EBS), except for waste packages (WPs), which are discussed in Chapter 5. As defined in proposed 10 CFR 63.2 (64 FR 8640), EBS means the WPs plus the underground facility. On that basis, the Chapter 6 portion of the EBS is the underground facility. When the term EBS is used in Chapter 6 of the technical guidance document (TGD), it means underground facility. By following this TGD, the LA authors will produce a chapter that should support a determination with reasonable assurance by the U.S. Nuclear Regulatory Commission (NRC) that the underground facility can be safely operated. Because the EBS design has no licensing precedent, the level of detail required in Chapter 6 of the LA is expected to be more than that required for those chapters that contain structures, systems, and components (SSCs) that have licensing precedent. Authors preparing the LA shall use this document for guidance and must read the Introduction and Appendix B before writing their respective sections.

As noted in proposed 10 CFR 63.24 (64 FR 8640), all LA information may not be available at the time of docketing. Therefore, the guidance in this chapter specifies a differentiated approach for providing the required information. Information requirements identified as "Information Required at the Time of Construction Authorization" must be provided for that submittal. Requirements identified as "Information Required at the Time of Update to the License Appliction to Receive and Possess" must be provided at that time.

6. PURPOSE AND SUMMARY

State that the purpose of Chapter 6 is to provide a description of the EBS and to evaluate compliance with design criteria to support the radiological safety case and waste isolation. State that each section in this chapter addresses how the information contained therein relates to the demonstration of compliance with the performance objectives of proposed 10 CFR 63.111 and proposed 10 CFR 63.113 (64 FR 8640). State that in conformance with the postclosure performance objective of proposed 10 CFR 63.113 (64 FR 8640), the repository design includes multiple barriers (natural and engineered). State that Chapter 6 describes the engineered barriers. Refer to Section 2.2 for an overview of project conformance with proposed 10 CFR 63.113 (64 FR 8640). Refer also to Section 8.6 for additional information regarding multiple barriers and their analyses (proposed 10 CFR 63.113(a) [64 FR 8640]).

State that by regulatory definition in proposed 10 CFR 63.2 (64 FR 8640), the WP is a part of the EBS. Delineate the boundary between the WP, which is discussed in Chapter 5, and the remainder of the EBS, which is discussed in this chapter. Provide a reference to Sections 5.1 and 5.2 for discussion of the waste forms and design of the WPs.

Discuss the organization of Chapter 6. Summarize the role of the information presented in Chapter 6 as it relates to the ability of the underground facility to operate safely. Explain how the EBS design supports safe operation of the repository, as demonstrated in the safety case in Chapter 7. Describe, in general terms, the relationship between the EBS design, described in Chapter 6, and the repository preclosure radiological safety assessment and postclosure performance assessment (PA), which are described in Chapters 7 and 8, respectively.

6.1 ENGINEERED BARRIER SYSTEM OVERVIEW DESCRIPTION

The EBS consists of the underground facilities, described in Chapter 6, and the WPs, described in Chapter 5. The term "engineered barrier system" as used in Chapter 6 means the "underground facilities." The underground facilities consist of the underground structure, the backfill material, and the underground openings. Underground openings include ramps, shafts, and boreholes, plus any seals used on the openings.

The underground structure is described in Sections 6.2.4 through 6.2.11, and 6.3.4. The backfill material and design is described in Section 6.3.3. The underground openings and their seals are described in Section 6.2.2, particularly Sections 6.2.2.6 through 6.2.2.9, 6.2.3, 6.3.1, and 6.3.2.

Briefly describe, at a summary level, the design features of the EBS and its components and how they are effective against the release of radioactive material to the environment.

6.1.1 Engineered Barrier System Environment

Provide a general description of the expected EBS environment. Describe the relationship of the EBS environment to the overall repository performance.

Provide an overview of the major site features that affect the EBS environment. Describe both the pre-emplacement and post-emplacement environments. Focus on the interactions of the environment with the EBS and on the effects of the environment on the ability of the EBS components to meet design criteria. Include pertinent site conditions that constitute the environment, including ambient temperature; mechanical, physical, and chemical properties of the host rock; the geology of the site, such as fault and seismic information; water chemistry; and water flow rate. Provide a reference to Chapter 3 for a more detailed discussion on these topics.

6.1.2 Level of Design Detail

This section provides the guidance for the required level of detail that must be provided for the SSCs that are discussed in Chapter 6. There will not be a Section 6.1.2 in the LA. Note also that Section 6.1.1 will not exist in the LA because there will be no Section 6.1.2. The information required by Section 6.1.1 of the TGD will be provided in Section 6.1 of the LA.

Provide the level of design detail given below for each of the EBS SSCs.

Quality Level (QL) 1, 2, and 3 SSCs and non-safety SSCs must be described in the LA. For each SSC, identify the QL. Provide a reference to Section 2.4 for more information related to QL classification. For any SSC, the level of design detail required is dependent on its:

- Importance to protection of the public health and safety
- Importance to protection of worker health and safety
- Need to demonstrate compliance with regulatory requirements
- Need to support submittal of a docketable LA.

The QL information following this paragraph lists the specific information that must be included in the description of SSCs falling into one of these four classifications. If a specific information

item is not relevant for a given SSC, that information item does not need to be addressed for that SSC in the LA.

6.1.2.1 Quality Level 1 Structures, Systems, and Components

The following information, as applicable, must be provided for each QL 1 SSC.

Regulatory Bases–These are primarily the proposed 10 CFR 63 (64 FR 8640) requirements.

- Identify which SSCs are relied upon to limit or prevent potential accidents or mitigate their consequences. Refer to Chapter 7 for details on the analyses that identified the SSCs as such. Identify and describe the function of the SSCs, including controls, relied upon to limit, prevent, or mitigate the consequences of design basis events (DBEs). Use drawings (e.g., piping and instrument diagrams, electrical one-line diagrams, general arrangement drawings, and handling drawings) as necessary to present the information for QL 1 SSCs. Include identification of measures taken to ensure the availability of identified safety systems (Interim Guidance Section 112(e) [Dyer and Horton 1999]). For each SSC identified in Chapter 7 as being affected by a Category 1 DBE, discuss the design considerations that prevent releases of radioactive materials that could result in an annual total effective dose equivalent (TEDE) of 0.25 mSv (25 mrem) to an individual member of the public beyond the boundary of the preclosure controlled area. Where appropriate, identify the sequence of events and how the system responds to the event (proposed 10 CFR 63.111(a)(2) and proposed 10 CFR 63.111(b)(1) [64 FR 8640]). Identify the SSCs that include design considerations until permanent closure to limit the consequences of Category 2 DBEs. Identify the specific DBEs and reference Chapter 7 for the analysis that identifies these events and the SSCs that are involved in preventing or mitigating the consequences of the events. Describe the design features incorporated into the SSC and its function that prevents or mitigates the consequences of the Category 2 DBEs (Interim Guidance Section 111(b)(2) [Dyer and Horton 1999]).

- Identify SSCs that limit and control radiation exposures and radiation levels in both restricted and unrestricted areas, and releases of radioactive materials to unrestricted areas from exceeding the limits specified in 10 CFR 20.1201 and 10 CFR 20.1301, and the as low as is reasonably achievable (ALARA) provisions of 10 CFR 20.1101(b) and 10 CFR 20.1101(d). The limits are as follows:

  - No worker shall receive the more limiting TEDE of 0.05 Sv (5 rem), or the sum of deep-dose equivalent and committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.50 Sv (50 rem) annually. The annual dose equivalent to the lens of the eye shall not exceed 0.15 Sv (15 rem). The annual shallow dose equivalent to the skin or to any extremity shall not exceed 0.50 Sv (50 rem).

  - No individual member of the public shall receive a TEDE in excess of 1 mSv (100 mrem) annually, and the dose in any unrestricted area from external sources does not exceed 0.02 mSv (2 mrem) hourly (proposed 10 CFR 63.21(c)(14), proposed 10 CFR 63.111(a)(1), and proposed 10 CFR 63.111(b)(1) [64 FR 8640]).

- Identify ALARA design considerations for facility features that limit and control occupational dose and dose to members of the public as required by 10 CFR 20.1101(b) and 10 CFR 20.1101(d) (proposed 10 CFR 63.21(c)(14) and proposed 10 CFR 63.111(a)(1) [64 FR 8640]).

- Identify and describe the design considerations for systems that monitor and control effluents. Describe design considerations for facility features and systems that control and monitor radiation levels to limit occupational radiation exposure (proposed 10 CFR 63.21(c)(14) and proposed 10 CFR 63.111(a)(1) [64 FR 8640]).

License Specifications

- **Information Required at the Time of Construction Authorization**–Identify when an item of equipment or a parameter is being considered as the subject of a license specification. Only equipment and parameters that are credited in the safety analysis with mitigating the consequences of Category 1 or 2 DBEs will be the subject of license specifications. Include the supporting information that will demonstrate how and why the equipment or parameter is credited with mitigating the consequences of a Category 1 or 2 DBE. Include a reference to Chapter 11, which will identify the license specifications and summarize the basis for selection of each item as a potential subject for a license specification (proposed 10 CFR 63.21(c)(13) [64 FR 8640]).

- **Information Required at the Time of Update to the License Application to Receive and Possess**–Identify the items that have been retained from the construction authorization (CA) stage as candidates for license specifications. Include the supporting information, as at the CA stage, to demonstrate how and why each candidate item is credited in the safety analysis with mitigating the consequences of Category 1 or 2 DBEs. Include the reference to Chapter 11 for the discussion of the license specification development process (proposed 10 CFR 63.21(c)(13) [64 FR 8640]).

Applicable Codes and Standards–List the codes and standards (including guidance documents and technical positions) required in the design and construction of the QL 1 SSCs. List the codes and standards on a structure or system level. Obtain the codes and standards from the applicable system description documents (SDDs) (proposed 10 CFR 63.21(c)(3) [64 FR 8640]).

Design Criteria–Identify the principal design criteria for each QL 1 SSC discussed in Chapter 6. The principal design criteria for a given SSC are found in Section 1 of the SDD for that SSC. Design criteria are defined as "standards or rules against which a design can be judged" or as "criteria that establish the necessary design, fabrication, construction, and performance requirements for SSCs." For each principal design criterion identified, include a discussion of the relationship of that criterion to the performance objective specified in proposed 10 CFR 63.113(b) (64 FR 8640). This objective states that the EBS must be designed so that the engineered barriers, together with the natural barriers, will limit the annual dose resulting from radioactive material released from the repository to the average member of the critical group to a maximum of 0.25 mSv (25 mrem) during the first 10,000 years after permanent closure (proposed 10 CFR 63.21(c)(4)(i) [64 FR 8640]).

Design Bases–Identify and discuss the design bases and their relationship to the principal design criteria for the QL 1 EBS SSCs. The design bases are the system functions found in Section 1.1 of the applicable SDDs plus the performance parameters found in Sections 1.2.1 through 1.2.5 of the SDDs. This definition is consistent with that of "design bases" found in proposed 10 CFR 63.2 (64 FR 8640). The design criteria are also found in Sections 1.2.1 through 1.2.5 of the SDDs. Base the discussion on the SDD criteria basis statements found in Section 5 of the SDDs (proposed 10 CFR 63.21(c)(4)(ii) [64 FR 8640].

General Description–Describe the QL 1 EBS SSCs, equipment, and process activities. Base the description on the information contained in Section 1 of related SDDs. Include the information required to support the safety analysis of the system or information that can be readily derived from it. Summarize the system functions, operations, the system design, concept of operations, and a description of system interfaces. Include a discussion of any special construction or fabrication techniques, unique testing programs, or special design and analysis procedures used for the SSCs, as applicable. For these SSCs, include drawings that show information needed to support the safety case. Other drawings and diagrams are not required, although they may be included to show concepts or ideas as needed to support the text (proposed 10 CFR 63.112(a) [64 FR 8640]).

Identify and discuss the materials of construction for the QL 1 EBS SSCs. Include general arrangement information and approximate dimensions for these materials (proposed 10 CFR 63.21(c)(3) [64 FR 8640]).

- **Information Required at the Time of Construction Authorization**–Include only the principal materials used in the design of SSCs that either prevent or mitigate a Category 1 or 2 DBE or are required for worker safety. Examples of the principal materials to be discussed include structural materials (e.g., steel beams, reinforced concrete), system component materials (e.g., stainless steel piping, air filters with charcoal absorbers), and shielding materials (e.g., leaded glass). Discuss the materials of construction on an SSC level. Include a general description of the geologic media (including general arrangement and dimensional information) that host the subsurface facilities, with reference to the appropriate sections in Chapter 3 for details (proposed 10 CFR 63.21(c)(3) [64 FR 8640]).

- **Information Required at the Time of Update to the License Application to Receive and Possess**–In addition to what is provided at the time of CA, present a detailed description of the materials of construction on a component basis. The materials are discussed in the description of the specific SSC. The information should include material compatibility with the environment during normal operations or post-accident situations, whichever is the most limiting, and as it relates to the service that the component provides. Examples from an existing licensed facility are provided below:

  - High-efficiency particulate air (HEPA) filters

    HEPA filter elements are made of pleated fiberglass with an aluminum separator design, measure 24 x 24 x 11.5 in., and are each capable of handling a nominal flow

rate of 1,000 ft^3/min. The filter medium is cased in stainless steel, has face guards on both sides, and is water- and fire-resistant.

– Cooling coils

The cooling coils are of nonferrous construction with aluminum fins mechanically bonded to seamless copper tubing. Coils are arranged for counter-flow operation using chilled water. The tube bundle is enclosed in a steel frame.

– Low total dissolved solids holdup tank (T-01 C)

Quantity per unit	= 1
Capacity (each)	= 30,000 gal
Design pressure and temperature	= Atmospheric pressure and 150°F
Operating pressure and temperature	= Atmospheric pressure and 80°F
Material	= 304 stainless steel

(proposed 10 CFR 63.21(c)(3) [64 FR 8640])

Research and Development

- **Information Required at the Time of Construction Authorization**–Identify the QL 1 SSCs that require research and development (R&D) to confirm the adequacy of design. Provide available information that describes the type of R&D required and the reason the additional information is needed. Refer to Chapter 11 for a description of the R&D program and the associated schedule (proposed 10 CFR 63.21(c)(21) [64 FR 8640]).

- **Information Required at the Time of Update to the License Application to Receive and Possess**–Provide the results from the required R&D identified in the LA at the time of CA for the various SSCs. Ensure all design chapter requirements for additional R&D are addressed and include the reasons that the additional information was originally needed (proposed 10 CFR 63.21(c)(21) [64 FR 8640]).

Drawings and Diagrams–The drawings and diagrams required for the QL 1 SSCs discussed in Chapter 6 are those that show information to support the safety case. Typical types of drawings and diagrams are listed below (proposed 10 CFR 63.112(a) [64 FR 8640]).

- Piping and instrument diagrams
- Electrical one-line diagrams
- General arrangement drawings
- Handling diagrams.

6.1.2.2 Quality Level 2 Structures, Systems, and Components

The following information, as applicable, must be provided for each QL 2 SSC.

Regulatory Bases–These are primarily the proposed 10 CFR 63 (64 FR 8640) requirements.

- Identify which SSCs are relied upon to limit or prevent potential accidents or mitigate their consequences, and refer to Chapter 7 for details on the analyses that identified the SSCs as such. Identify and describe the function of the SSCs, including controls, relied upon to limit, prevent, or mitigate the consequences of DBEs. Include identification of measures taken to ensure the availability of identified safety systems (Interim Guidance Section 112(e) [Dyer and Horton 1999]). For each QL 2 SSC identified in Chapter 7 as being affected by a Category 1 DBE, discuss the design considerations that prevent releases of radioactive materials that could result in an annual TEDE of 0.25 mSv (25 mrem) to an individual member of the public beyond the boundary of the preclosure controlled area. Where appropriate, identify the sequence of events and how the system responds to the event (Interim Guidance Section 111(a)(2) [Dyer and Horton 1999] and proposed 10 CFR 63.111(b)(1) [64 FR 8640]). Identify the SSCs that include design considerations until permanent closure to limit the consequences of Category 2 DBEs. Identify the specific DBEs and reference Chapter 7 for the analysis that identifies these events and the SSCs that are involved in preventing or mitigating the consequences of the events. Describe the design features incorporated into the QL 2 SSC and its function that prevents or mitigates the consequences of Category 2 DBEs (Interim Guidance Section 111(b)(2) [Dyer and Horton 1999].

- Identify SSCs that limit and control radiation exposures and radiation levels in both restricted and unrestricted areas and releases of radioactive materials to unrestricted areas from exceeding the limits specified in 10 CFR 20.1201, 10 CFR 20.1301, and the ALARA provisions of 10 CFR 20.1101(b) and 10 CFR 20.1101(d). The limits are as follows:

  - No worker shall receive the more limiting TEDE of 0.05 Sv (5 rem), or the sum of deep-dose equivalent and committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.50 Sv (50 rem), annually. The annual dose equivalent to the lens of the eye shall not exceed 0.15 Sv (15 rem). The annual shallow dose equivalent to the skin or to any extremity shall not exceed 0.50 Sv (50 rem).

  - No individual member of the public shall receive a TEDE in excess of 1 mSv (100 mrem) annually, and the dose in any unrestricted area from external sources does not exceed 0.02 mSv (2 mrem) hourly (proposed 10 CFR 63.21(c)(14), proposed 10 CFR 63.111(a)(1), and proposed 10 CFR 63.111(b)(1) [64 FR 8640]).

- Identify ALARA design considerations for facility features that limit and control occupational dose and dose to members of the public as required by 10 CFR 20.1101(b) and 10 CFR 20.1101(d) (proposed 10 CFR 63.21(c)(14) and proposed 10 CFR 63.111(a)(1) [64 FR 8640]).

- Describe the design considerations for systems that monitor and control effluents. Describe design considerations for facility features and systems that control and monitor radiation levels to limit occupational radiation exposure (proposed 10 CFR 63.21(c)(14) and proposed 10 CFR 63.111(a)(1) [64 FR 8640]).

License Specifications

- **Information Required at the Time of Construction Authorization**–Identify when an item of equipment or a parameter is being considered as the subject of a license specification. Only equipment and parameters that are credited in the safety analysis with mitigating the consequences of Category 1 or 2 DBEs will be the subject of license specifications. Include the supporting information that will demonstrate how and why the equipment or parameter is credited with mitigating the consequences of a Category 1 or 2 DBE. Include a reference to Chapter 11, which will identify the license specifications and summarize the basis for selection of each item as a potential subject for a license specification (proposed 10 CFR 63.21(c)(13) [64 FR 8640]).

- **Information Required at the Time of Update to the License Application to Receive and Possess**–Identify the items that have been retained from the CA stage as candidates for license specifications. Include the supporting information, as at the CA stage, to demonstrate how and why each candidate item is credited in the safety analysis with mitigating the consequences of Category 1 or 2 DBEs. Include a reference to Chapter 11 for the discussion of the license specification development process (proposed 10 CFR 63.21(c)(13) [64 FR 8640]).

Applicable Codes and Standards–List the codes and standards (including guidance documents and technical positions) required in the design and construction of the QL 2 SSCs. List the codes and standards on a structure or system level. Obtain the codes and standards from the applicable SDDs (proposed 10 CFR 63.21(c)(3) [64 FR 8640]).

Design Criteria–Identify the principal design criteria for each QL 2 SSC discussed in Chapter 6. The principal design criteria for a given SSC are found in Section 1 of the SDD for that SSC. Design criteria are defined as "standards or rules against which a design can be judged" or as "criteria that establish the necessary design, fabrication, construction, and performance requirements for SSCs." For each principal design criterion identified, include a discussion of the relationship of that criterion to the performance objective specified in proposed 10 CFR 63.113(b) (64 FR 8640). This objective states that the EBS must be designed so that, together with the natural barriers, the annual dose resulting from radioactive material released from the repository to the average member of the critical group does not exceed 0.25 mSv (25 mrem) during the first 10,000 years after permanent closure (proposed 10 CFR 63.21(c)(4)(i) [64 FR 8640]).

General Description–Describe the SSCs, equipment, and process activities. Base the description on the information contained in Section 1 of related SDDs. Include the information required to support the safety analysis of the system or information that can be readily derived from it. Summarize the system functions, operations, the system design, concept of operations, and a description of system interfaces. Include a discussion of any special construction or

fabrication techniques, unique testing programs, or special design and analysis procedures used for the SSCs, as applicable (proposed 10 CFR 63.112(a) [64 FR 8640]).

Identify and discuss the materials of construction for the QL 2 EBS SSCs. Include general arrangement information and approximate dimensions for these materials (proposed 10 CFR 63.21(c)(3) [64 FR 8640]).

- **Information Required at the Time of Construction Authorization**–Include only the principal materials used in the design of SSCs that either prevent or mitigate a Category 1 or 2 DBE or are required for worker safety. Examples of the principal materials to be discussed include structural materials (e.g., steel beams, reinforced concrete), system component materials (e.g., stainless steel piping, air filters with charcoal absorbers), and shielding materials (e.g., leaded glass). Discuss the materials of construction on an SSC level. Include a general description of the geologic media (including general arrangement and dimensional information) that host the subsurface facilities, with reference to the appropriate sections in Chapter 3 for details (proposed 10 CFR 63.21(c)(3) [64 FR 8640]).

- **Information Required at the Time of Update to the License Application to Receive and Possess**–In addition to what is provided at the time of CA, present a detailed description of the materials of construction on a component basis. The materials are discussed in the description of the specific SSC. The information should include material compatibility with the environment during normal operations or post-accident situations, whichever is the most limiting, and as it relates to the service that the component provides. Examples from an existing licensed facility are provided below:

  – HEPA filters

    HEPA filter elements are made of pleated fiberglass with an aluminum separator design, measure 24 x 24 x 11.5 in., and are each capable of handling a nominal flow rate of 1,000 ft^3/min. The filter medium is cased in stainless steel, has face guards on both sides, and is water- and fire-resistant.

  – Cooling coils

    Cooling coils are of nonferrous construction with aluminum fins mechanically bonded to seamless copper tubing. Coils are arranged for counter-flow operation using chilled water. The tube bundle is enclosed in a steel frame.

– Low total dissolved solids holdup tank (T-01 C)

Quantity per unit	= 1
Capacity (each)	= 30,000 gal
Design pressure and temperature	= Atmospheric pressure and 150°F
Operating pressure and temperature	= Atmospheric pressure and 80°F
Material	= 304 stainless steel

(proposed 10 CFR 63.21(c)(3) [64 FR 8640]).

Research and Development

- **Information Required at the Time of Construction Authorization**–Identify the QL 2 SSCs that require R&D to confirm the adequacy of design. Provide available information that describes the type of R&D required and the reason the additional information is needed. Refer to Chapter 11 for a description of the R&D program and the associated schedule (proposed 10 CFR 63.21(c)(21) [64 FR 8640]).

- **Information Required at the Time of Update to the License Application to Receive and Possess**–Provide the results from the required R&D identified in the LA at the time of CA for the various SSCs. Ensure all design chapter requirements for additional R&D are addressed, and include the reasons that the additional information was originally needed (proposed 10 CFR 63.21(c)(21) [64 FR 8640]).

6.1.2.3 Quality Level 3 Structures, Systems, and Components

The following information, as applicable, must be provided for each QL 3 SSC.

Regulatory Bases–These are primarily proposed 10 CFR 63 (64 FR 8640) requirements.

- Identify which QL 3 SSCs are relied upon to limit or prevent potential accidents or mitigate their consequences, and refer to Chapter 7 for details on the analyses that identified the SSCs as such. Identify and describe the function of the SSCs, including controls, relied upon to limit, prevent, or mitigate the consequences of DBEs. Include identification of measures taken to ensure the availability of identified safety systems (Interim Guidance Section 112(e) [Dyer and Horton 1999]). For each QL 3 SSC identified in Chapter 7 as being affected by a Category 1 DBE, discuss the design considerations that prevent releases of radioactive materials that could result in an annual TEDE of 0.25 mSv (25 mrem) to an individual member of the public at the boundary. Where appropriate, identify the sequence of events and how the system responds to the event (Interim Guidance Section 111(a)(2) [Dyer and Horton 1999] and proposed 10 CFR 63.111(b)(1) [64 FR 8640]). Identify the SSCs that include design considerations until permanent closure to limit the consequences of Category 2 DBEs. Identify the specific DBEs and reference Chapter 7 for the analysis that identifies these events and the SSCs that are involved in preventing or mitigating the consequences of the events. Describe the design features incorporated into the QL 3 SSC and its function

that prevents or mitigates the consequences of Category 2 DBEs (Interim Guidance Section 111(b)(2) [Dyer and Horton 1999]).

- Identify any QL SSCs that limit and control radiation exposures and radiation levels in both restricted and unrestricted areas and releases of radioactive materials to unrestricted areas from exceeding the limits specified in 10 CFR 20.1201, 10 CFR 20.1301, and the ALARA provisions of 10 CFR 20.1101(b) and 10 CFR 20.1101(d). The limits are as follows:

  - No worker shall receive the more limiting TEDE of 0.05 Sv (5 rem), or the sum of deep-dose equivalent and committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.50 Sv (50 rem) annually. The annual dose equivalent to the lens of the eye shall not exceed 0.15 Sv (15 rem). The annual shallow dose equivalent to the skin or to any extremity shall not exceed 0.50 Sv (50 rem).

  - No individual member of the public shall receive a TEDE in excess of 1 mSv (100 mrem) annually, and the dose in any unrestricted area from external sources does not exceed 0.02 mSv (2 mrem) hourly (proposed 10 CFR 63.21(c)(14), proposed 10 CFR 63.111(a)(1), and proposed 10 CFR 63.111(b)(1) [64 FR 8640]).

- Identify ALARA design considerations for any QL 3 facility features that limit and control occupational dose and dose to members of the public as required by 10 CFR 20.1101(b) and 10 CFR 20.1101(d) (proposed 10 CFR 63.21(c)(14) and proposed 10 CFR 63.111(a)(1) [64 FR 8640]).

- Identify and describe the design considerations for any QL 3 systems that monitor and control effluents. Describe design considerations for facility features and systems that control and monitor radiation levels to limit occupational radiation exposure (proposed 10 CFR 63.21(c)(14) and proposed 10 CFR 63.111(a)(1) [64 FR 8640]).

License Specifications

- **Information Required at the Time of Construction Authorization**–Identify when any QL 3 item of equipment or a related parameter is being considered as the subject of a license specification. Only equipment and parameters that are credited in the safety analysis with mitigating the consequences of Category 1 or 2 DBEs will be the subject of license specifications. Include the supporting information that will demonstrate how and why the equipment or parameter is credited with mitigating the consequences of a Category 1 or 2 DBE. Include a reference to Chapter 11, which will identify the license specifications and summarize the basis for selection of each item as a potential subject for a license specification (proposed 10 CFR 63.21(c)(13) [64 FR 8640]).

- **Information Required at the Time of Update to the License Application to Receive and Possess**–Identify any QL 3 items that have been retained from the CA stage as candidates for license specifications. Include the supporting information, as at the CA stage, to demonstrate how and why each candidate item is credited in the safety analysis

with mitigating the consequences of Category 1 or 2 DBEs. Include a reference to Chapter 11 for the discussion of the license specification development process (proposed 10 CFR 63.21(c)(13) [64 FR 8640]).

Design Criteria–Identify the principal design criteria for each QL 3 SSC discussed in Chapter 6. The principal design criteria for a given SSC are found in Section 1 of the SDD for that SSC. Design criteria are defined as "standards or rules against which a design can be judged" or as "criteria that establish the necessary design, fabrication, construction, and performance requirements for SSCs." For each principal design criterion identified, include a discussion of the relationship of that criterion to the performance objective specified in proposed 10 CFR 63.113(b) (64 FR 8640). This objective states that the EBS must be designed so that the engineered barriers, together with the natural barriers, limit the annual dose resulting from radioactive material released from the repository to the average member of the critical group to a maximum of 0.25 mSv (25 mrem) during the first 10,000 years after permanent closure (proposed 10 CFR 63.21(c)(4)(i) [64 FR 8640]).

General Description–Describe the SSCs, equipment, and process activities. Base the description on the information contained in Section 1 of the related SDDs. Include the information required to support the safety analysis of the system or information that can be readily derived from it. Summarize the system functions, operations, the system design, concept of operations, and a description of system interfaces. Include a discussion of any special construction or fabrication techniques, unique testing programs, or special design and analysis procedures used for the SSCs, as applicable (proposed 10 CFR 63.112(a) [64 FR 8640]).

Identify and discuss the materials of construction for the QL 3 EBS SSCs. Include general arrangement information and approximate dimensions for these materials (proposed 10 CFR 63.21(c)(3) [64 FR 8640]).

- **Information Required at the Time of Construction Authorization**–Include only the principal materials used in the design of SSCs that either prevent or mitigate a Category 1 or 2 DBE or are required for worker safety. Examples of the principal materials to be discussed include structural materials (e.g., steel beams, reinforced concrete), system component materials (e.g., stainless steel piping, air filters with charcoal absorbers), and shielding materials (e.g., leaded glass). Discuss the materials of construction on an SSC level. Include a general description of the geologic media (including general arrangement and dimensional information) that host the subsurface facilities, with reference to the appropriate sections in Chapter 3 for details (proposed 10 CFR 63.21(c)(3) [64 FR 8640]).

- **Information Required at the Time of Update to the License Application to Receive and Possess**–In addition to what is provided at the time of CA, a detailed description of the materials of construction must be presented. This should be done on a component basis as the materials are discussed in the description of the specific SSC. The information should include material compatibility with the environment during normal operations or post-accident situations, whichever is the most limiting, and as it relates to the service that the component provides. Examples from an existing licensed facility are provided below:

– HEPA filters

HEPA filter elements are made of pleated fiberglass with an aluminum separator design, measure 24 x 24 x 11.5 in., and are each capable of handling a nominal flow rate of 1,000 ft^3/min. The filter medium is cased in stainless steel, has face guards on both sides, and is water- and fire-resistant.

– Cooling coils

Cooling coils are of nonferrous construction with aluminum fins mechanically bonded to seamless copper tubing. Coils are arranged for counter-flow operation using chilled water. The tube bundle is enclosed in a steel frame.

– Low total dissolved solids holdup tank (T-01 C)

Quantity per unit	= 1
Capacity (each)	= 30,000 gal
Design pressure and temperature	= Atmospheric pressure and 150°F
Operating pressure and temperature	= Atmospheric pressure and 80°F
Material	= 304 stainless steel

(proposed 10 CFR 63.21(c)(3) [64 FR 8640]).

Research and Development

- **Information Required at the Time of Construction Application**–Identify the QL 3 SSCs that require R&D to confirm the adequacy of design. Provide available information that describes the type of R&D required and the reason the additional information is needed. Refer to Chapter 11 for a description of the R&D program and the associated schedule (proposed 10 CFR 63.21(c)(21) [64 FR 8640]).

- **Information Required at the Time of Update to the License Authorization to Receive and Possess**–Provide the results from the required R&D identified in the LA at the time of CA for the various SSCs. Ensure all design chapter requirements for additional R&D are addressed, and include the reasons that the additional information was originally needed (proposed 10 CFR 63.21(c)(21)[64 FR 8640]).

6.1.2.4 Non-Safety Structures, Systems, and Components

The information that must be provided for any non-safety SSCs discussed in this chapter includes:

Regulatory Bases–These are primarily the proposed 10 CFR 63 (64 FR 8640) requirements.

General Description–Provide a general description to the extent that is sufficient to demonstrate the non-safety classification. Base the general description on information contained in Section 1 of the applicable SDDs. Drawings and diagrams may be included to show concepts or ideas to support the text to the extent needed to demonstrate the non-safety classification.

6.2 ENGINEERED BARRIER SYSTEM DESIGN AND OPERATION

Provide an overview of the underground facilities, providing the required level of design detail for any EBS SSCs that are not described elsewhere in Chapter 6, as outlined in Section 6.1.2.

6.2.1 Thermal Management

Describe the design features of the repository that help to ensure that the spent nuclear fuel and high-level radioactive waste material temperatures do not exceed their allowable values during normal, off-normal, and accident conditions and, thus, contribute to meeting the overall repository performance objectives. Describe the purpose of this section as it relates to overall repository performance. Show that the resulting heat generated by the waste when emplaced according to the selected design will be acceptable from a performance perspective. Include a discussion of the rock temperature limits and the design measures that will ensure the limits are not exceeded. Provide a reference to Chapter 8 for performance results.

6.2.1.1 Design Bases

Describe the design bases of the EBS SSCs that are discussed in Section 6.2.1, or refer to where they are described elsewhere in Chapter 6.

6.2.1.2 Design Description

Describe the thermal management features of the repository.

6.2.1.3 Design Evaluation

Discuss the analyses performed that provide reasonable assurance of compliance with applicable design requirements related to thermal loading. Also, describe the interface between the design evaluation described in this section and the preclosure radiological safety and postclosure overall repository performance described in Chapters 7 and 8, respectively.

State that collection, documentation, and development of all data, methods, models, or computer codes have been performed in accordance with the *Quality Assurance Requirements and Description* (DOE 1998) and associated implementing procedures for any thermal-mechanical analyses regarding design of the underground facility or flow into the drifts not covered in Chapter 8 [RDTME 3.1.1] [RDTME 3.3.1] (NRC 1998a).

Discuss and provide examples of how the thermal-mechanical design makes use of site-specific thermal and mechanical properties. Show that the spatial distribution of such properties is implemented in the EBS thermal-mechanical analyses for the design [RDTME 3.1.3] (NRC 1998a).

Show that the process to develop inputs to the thermal-mechanical design includes consideration of associated uncertainties. Either identify or refer to a document that identifies the potential impacts of the uncertainties on the design [RDTME 3.1.4] (NRC 1998a).

Describe time-dependent changes in the size and shape of drifts due to thermally induced ground movement (e.g., rock deformations, rock collapse, and other changes that may affect the integrity and geometric configuration of the drifts) and discuss how they are considered in the assessment of flow into the emplacement drifts. Show how magnitude and distribution of the drift geometry changes are consistent with the results of the thermal-mechanical analyses of the underground facility [RDTME 3.3.3] (NRC 1998a).

Describe how the results of the thermal-mechanical analyses, including consideration of ground support stability, are accounted for in the underground facility maintenance requirements [RDTME 3.1.16] (NRC 1998a).

Describe changes in the hydrological properties of the rock (e.g., fracture porosity or permeability) due to thermally induced ground movement, and discuss how they are considered in the analyses assessing flow into the emplacement drifts. Show how the magnitude and distribution of the drift geometry changes are consistent with the results of the thermal-mechanical analyses of the underground facility [RDTME 3.3.4] (NRC 1998a).

Describe any monitoring and mechanical testing planned for the EBS SSCs during the performance confirmation period to confirm that they will function as intended when subject to thermal and stress perturbations [CLST 2.7] (NRC 1998b).

6.2.2 Subsurface Description

Provide a brief description of the subsurface design of the EBS except for the WP, which is described in Chapter 5. Discuss major EBS components and their intended functions and relationships to the overall repository design.

State that expert elicitations are conducted and documented in accordance with the requirements of the *Quality Assurance Requirements and Description* (DOE 1998). Identify any expert elicitations that have been used in support of design of underground facilities or assessments of flow into emplacement drifts and the extent to which each was used [RDTME 3.1] [RDTME 3.3] (NRC 1998a).

Discuss specific aspects of the design, such as materials, thermal, structural, criticality, and shielding, in the following subsections.

6.2.2.1 Engineered Barrier System Materials Design

Provide the required level of design detail, as outlined in Section 6.1.2, for the EBS materials. Provide supplemental information as required by the guidance given below.

Include a summary of any ground support materials (e.g., concrete) that have been added to the EBS environment. Discuss the impact of these materials on radionuclide solubility, radionuclide transport, host rock permeability, and repository performance.

6.2.2.2 Engineered Barrier System Thermal Design

Describe aspects of the EBS design and its components related to thermal performance.

6.2.2.3 Engineered Barrier System Criticality Design

Identify any EBS SSCs that are subject to criticality design criteria, criticality design bases, or other criticality design considerations.

Describe how the EBS design is expected to support the repository to meet its performance objectives regarding criticality.

6.2.2.4 Engineered Barrier System Structural Design

Provide the required level of design detail, as outlined in Section 6.1.2, for the EBS structural SSCs. Provide supplemental information as required by the guidance given below.

Show that seismic and fault-displacement data inputs for the EBS thermal-mechanical design are consistent with the topical report addressing design and PA inputs [RDTME 3.1.5] (NRC 1998a).

Discuss how the thermal-mechanical design and analyses make use of constitutive models that represent jointed rock mass behavior [RDTME 3.1.6] (NRC 1998a).

Discuss how both drift-scale and repository-scale models of the underground facility were used for the thermal-mechanical analyses to establish the intensity and distribution of ground movement, including rock deformation, rock collapse, and other changes that may affect the integrity or geometrical configuration of the underground openings [RDTME 3.1.8] (NRC 1998a).

6.2.2.5 Engineered Barrier System Shielding Design

Describe the shielding design for the underground facilities. Identify the shielding materials used, and provide the appropriate material properties. Include the level of detail required by Section 6.1.2 for the shielding design description.

6.2.2.6 Access Ramps

Provide the required level of design detail for the access ramps of the EBS, as outlined in Section 6.1.2. Provide supplemental information as required by the guidance given below.

North Ramp–Provide a brief introduction for the waste ramp, listing its primary function, history, and location. Describe how the North Ramp temporary ground support will be either replaced with permanent ground support or retained and qualified to become the permanent support.

State that the North Ramp is the primary access for transportation of WPs to the subsurface from the waste handling building. State that the North Ramp also serves as the primary corridor for the transportation of personnel, equipment, and materials to support emplacement operations, as well as serving as the primary intake airway for the emplacement side.

Provide the azimuth and gradient for the North Ramp. Describe its intersection with the East Main, the North Ramp Extension, and the East Main North Extension.

Discuss construction of the ramp lining and invert.

Describe the waste ramp operations. Briefly discuss the rail and overhead electric trolley system, the rail cars, and the electric-powered locomotives used to transport the WPs. Refer to Section 6.2.5 for further discussion of the waste emplacement system.

Discuss the use of the North Ramp as the primary intake airway for the emplacement side of the repository. Refer to Section 6.2.4 for additional details concerning ventilation.

South Ramp–Provide a brief introduction for the South Ramp, listing its primary functions and location.

Discuss the use of the South Ramp as the muck handling route, the location for the primary exhaust airway on the development side, and the transportation corridor for personnel, materials, and equipment. Discuss safety considerations in relation to potential fire hazards and available escape routes.

Show the connection to the South Ramp Extension and East Main. Provide the South Ramp azimuth and gradient.

State that the South Ramp serves as the primary exhaust airway for the development side of the repository. Provide a table showing representative airflow during repository operations. Refer to Section 6.2.4 for additional details concerning ventilation.

Discuss the personnel safety aspects of the integrated rail transport system for moving personnel and equipment in and out of the South Ramp.

6.2.2.7 Development Ventilation Shaft

Provide a brief introduction for the ventilation intake shaft for the subsurface development (excavation) side of the repository, listing its primary functions and location.

Provide the required level of design detail, as outlined in Section 6.1.2, for the ventilation intake shaft portion of the EBS. Provide supplemental information as required by the guidance given below.

State that the development side shaft is the primary ventilation intake airway and indicate whether it is equipped for personnel and material hoisting.

Describe the location of the ventilation intake shaft. Refer to a figure showing its location relative to the surface and underground.

Discuss the configuration of the development side intake shaft.

Discuss the function of the development side shaft (e.g., that it serves as an intake airway for the development side of the repository). Refer to Section 6.2.4 for additional details concerning ventilation.

6.2.2.8 Emplacement Ventilation Shaft

Provide a brief introduction for the ventilation exhaust shaft for the subsurface emplacement side of the repository, listing its primary functions and location.

Provide the required level of design detail, as outlined in Section 6.1.2, for the ventilation exhaust shaft portion of the EBS. Provide supplemental information as required by the guidance given below.

State that the emplacement side shaft is the primary exhaust airway and indicate whether it is equipped for personnel and material hoisting.

Describe the location, layout, and general arrangement of the ventilation exhaust shaft. Provide, or refer to, a figure showing the location of the exhaust shaft.

Discuss the configuration of the shaft and collar.

State that the emplacement side shaft functions as the only ventilation exhaust airway on the emplacement side of the repository.

6.2.2.9 Other Ramps and Shafts

Provide overview information on other ramps and shafts (including boreholes), if any, that penetrate the geologic repository operations area subsurface. Include information such as location, depth, diameter, lining, and purpose. For boreholes within the repository, provide the standoff distance.

Provide the required level of design detail, as outlined in Section 6.1.2, for the other ramps and shafts portion of the EBS. Provide supplemental information as required by the guidance given below.

Describe operations concepts for these other ramps, shafts, or boreholes, if any, and how they will be used initially; how they will be used during repository operations; the types of ramps, shafts, or boreholes that will be used; and the data that will be acquired.

6.2.3 Excavation and Ground Support Systems

Provide an overview description of the ground support systems and the basic equipment used for the excavated openings.

6.2.3.1 Excavation

Describe the mechanical excavation of the repository.

Provide a table containing the geomechanical data important to mechanical excavation. Refer to Section 3.2.7 for a description of the repository host horizon containing lithostratigraphic units and adjacent rock units. Describe the siting of the repository within the available repository siting volume.

6.2.3.2 Ground Support

Provide a description of the ground support system that will be used, including diagrams. Include emplacement drifts and non-emplacement openings (e.g., perimeter mains, turnouts, ramps, shafts). Provide a reference to the layout figure and other text that explains and depicts the layout.

Provide the required level of design detail, as outlined in Section 6.1.2, for the ground support SSCs. Provide supplemental information as required by the guidance given below.

Discuss the geotechnical data that provides input for ground support analysis and design, including borehole data and Exploratory Studies Facility (ESF) drift mapping data, ESF in situ testing data, and ESF construction and "lessons learned" information.

Specifically address the chemical impact of the materials used in the ground support system on waste isolation, radionuclide solubility, WP corrosion, and host rock permeability.

State that the preclosure seismic design method is documented in the Topical Report: *Preclosure Seismic Design Methodology for a Geologic Repository at Yucca Mountain* (YMP 1996). Clarify the differences in seismic design for the underground facilities and openings. Present preclosure underground seismic design methods in the context of designing for a 150-year maintainable service life for underground openings and ground support systems. State that the approach to fault displacement design is to avoid major faults and to provide sufficient stand-off distance between SSCs and the faults whenever possible.

Describe how the underground facility maintenance plans address stability of the underground openings, with particular attention to retrieval [RDTME 3.1.17] (NRC 1998a). Refer to Section 11.12 for a description of the retrieval plan.

Show that the seismic hazard inputs used to estimate rockfall potential are consistent with the inputs used in the design and PAs as established in the topical report addressing design and PA inputs [RDTME 3.2.3] (NRC 1998a).

Discuss how the size distribution of rocks that may fall onto the WPs is estimated. Show that the distribution is estimated from site-specific data, such as the distribution of joint patterns, spacing, and three-dimensional orientation. Discuss how uncertainties are considered [RDTME 3.2.4] (NRC 1998a).

Describe how time-dependent jointed rock behavior is considered in the thermal-mechanical analyses that provide the background conditions for the EBS seismic analysis [RDTME 3.2.6] (NRC 1998a).

Describe how the EBS rockfall dynamic analyses address the possibility of multiple rocks simultaneously falling onto a WP. Discuss how the extent of the potential rockfall area around an individual drift and around the entire repository is determined. Confirm that the area is determined as a function of ground motion [RDTME 3.2.7] (NRC 1998a).

6.2.4 Ventilation System

Introduce the ventilation system section by providing a short summary of its contents. Include the following three items in the summary:

- System capacity of the development and emplacement areas relative to the maximum ventilation requirements during the operational lifetime of the repository
- Fan characteristics, the ventilation control system, and ventilation monitoring
- Air cooling requirements for testing, maintenance, retrieval operations, and the number of days needed to cool the emplacement drifts for the design operating modes.

6.2.4.1 General Underground Ventilation Description

Describe the underground network requiring ventilation. Provide a figure showing general layout and airflow.

6.2.4.2 Development Ventilation System

Provide the required level of design detail, as outlined in Section 6.1.2, for this system. Provide supplemental information as required by the guidance given below.

Describe the ventilation flow for development air, referring to a figure that shows the flowpaths and control devices needed to distribute airflow in the development side.

6.2.4.3 Emplacement Ventilation System

Provide the required level of design detail, as outlined in Section 6.1.2, for this system. Provide supplemental information as required by the guidance given below.

State that ventilation of underground emplacement drifts is employed to allow monitoring of drift conditions and for removal of a portion of the waste heat generated during the preclosure period. Discuss the effect of ventilation on emplacement drift temperature, delaying the onset of the peak rock temperature and lowering the peak temperature when it does occur. Discuss the effect of ventilation on the removal of water vapor from the host rock mass during the post-emplacement period. Discuss the effect of the amount of water vapor given up by the rock mass on the thermal management process. Reference Section 6.2.1, as necessary.

Describe the design provisions for the abnormal condition of a loss of normal emplacement side ventilation.

Discuss the important parameters regarding air quantity determination for emplacement ventilation. Provide, or refer to, a figure that shows the flow paths and control devices needed to distribute the emplacement side airflow.

6.2.5 Waste Emplacement

Provide a summary of the contents of this section.

6.2.5.1 Waste Emplacement Equipment and Considerations

Provide the required level of design detail, as outlined in Section 6.1.2, for this system. Provide supplemental information as required by the guidance given below.

Discuss the waste emplacement concept. Provide figures showing the general emplacement configuration. Show the layout of emplacement drifts and the mode of emplacement.

Discuss waste handling equipment in the context of starting and final destinations, grades, travel distances, and operational transfers. Discuss design provisions for postulated accidents and breakdowns. Provide figures and tables containing design parameters (e.g., loads, environmental factors, radiation shielding considerations) for waste handling equipment for emplacement and for transport of WPs from surface facilities to subsurface emplacement locations.

6.2.5.2 Excavation and Ground Support of Emplacement Area

Describe the emplacement area excavation and ground support design. Provide a reference to Section 6.2.3 for additional information.

6.2.6 Waste Retrieval

Provide the required level of design detail, as outlined in Section 6.1.2, for the waste retrieval SSCs. Provide supplemental information as required by the guidance given below.

- Describe the design features of the underground facilities SSCs that will permit, enable, assist with, and participate in retrieval of any of the emplaced WPs during the specified retrieval period. The retrieval period starts when emplacement begins and continues for as long as 50 years, unless a different time period is established by the NRC (proposed 10 CFR 63.111(e)(1) [64 FR 8640]).

- Include a discussion that states that backfill can begin before the end of the period of design for retrievability. Discuss the basis needed for an early retrieval decision. Include a discussion that recognizes that the NRC may permanently close the repository before the end of the period of design for retrievability (proposed 10 CFR 63.111(e)(2) [64 FR 8640]).

Provide a reference to Section 6.2.4 for a discussion of the ventilation system requirements during retrieval.

Refer to Section 11.12 for a description and discussion of the retrieval plan (proposed 10 CFR 63.21(c)(19) [64 FR 8640]).

6.2.7 Emergency Systems

Identify the underground facility emergency systems. Describe the purpose of these emergency systems.

Provide the required level of design detail, as outlined in Section 6.1.2, for these systems.

6.2.8 Communication System

Describe the purpose for the subsurface communication system, which is to provide voice, video, and data links throughout the underground facilities and between the underground and surface facilities for both routine operations and emergencies.

Discuss the system and its functioning for both internal and external communications, with particular emphasis on the systems and SSCs that will be used under emergency conditions.

Provide the required level of design detail, as outlined in Section 6.1.2, for this system.

Provide a cross-reference to Section 1.6 for a discussion of safeguards and security operations. Provide a cross-reference to Section 11.11 for emergency planning considerations.

6.2.9 Operations Monitoring System

Describe the purpose and scope of the Operations Monitoring System

Provide the required level of detail, as outlined in Section 6.1.2, for this system.

6.2.10 Operations Support Systems

Describe the purpose for these systems, which is to provide support facilities and equipment for underground operations. List the individual systems included in this group of systems. Provide the required level of design detail, as outlined in Section 6.1.2, for these systems.

Consider the following systems for inclusion in this group:

- Mine wastewater drainage
- Lighting system
- Power systems
- Electrical systems
- Compressed air system
- Fuel supply
- Water supply
- Emergency services
- Auxiliary or back-up systems
- Maintenance shops
- Supply rooms
- Personnel showers
- Radiation monitoring

- Decontamination facilities
- Offices.

6.2.11 Other Underground Systems

Provide the purpose and description of other underground systems not covered elsewhere in Chapter 6 (e.g., drainage) that are part of the underground facility.

Provide the required level of design detail, as outlined in Section 6.1.2, for these systems.

List the individual systems included in this group of systems.

6.3 PERMANENT CLOSURE

Provide a summary of the permanent closure design provisions that affect the underground facility. Provide the required level of design detail, as outlined in Section 6.1.2, for the EBS SSCs that facilitate closure. Supplement that detail by addressing the information below.

- Identify the EBS SSCs that include design considerations to permit implementation of a performance confirmation program. Describe the design features incorporated into the SSC and its function that permits implementation of a performance confirmation program (proposed 10 CFR 63.111(d) [64 FR 8640]).

- Describe the design features and considerations that are intended to facilitate permanent closure of the underground facilities. Include short summary descriptions of the seals planned for the ramps, shafts, and boreholes (proposed 10 CFR 63.21(c)(18) [64 FR 8640]).

Identify the specific design features of the sealing and closure system that promote its long-term stability. Refer to Chapter 8 of the LA for discussion of long-term performance analysis and the extent to which these performance results are dependent upon the sealing system.

Discuss preparation of the subsurface facility for permanent closure. Discuss the long-term repository drainage and the long-term stability of emplacement drifts. Include any required linkages (interfaces) and associated interface boundaries with offsite systems or agencies. Describe any aspects of the WP corrosion testing program that are not covered in Chapters 5 or 8 [CLST 6.6] (NRC 1998b).

Describe any specific plans for further WP corrosion testing to reduce areas of uncertainty under the performance confirmation program that are not covered in Chapters 5, 8, or 12 [CLST 6.6] (NRC 1998b).

If the ceramic coating option is selected for WP design, describe any considerations used in the EBS analyses regarding the effect of the repository environment, mechanical impacts, other stresses, or corrosion on the ceramic coating integrity [CLST 6.2] (NRC 1998b). Describe and justify the use of the environmental conditions for any test results not specifically collected for the repository site [CLST 6.5] (NRC 1998b).

6.3.1 Seals

Describe the purpose and function of the EBS seals. Provide the required level of detail, as outlined in Section 6.1.2, for the postclosure seals portion of the EBS and as supplemented by the information below.

Describe the aspects of the seals design that help ensure that the seals will not become pathways for radioactive material release to the surface and will not create pathways for the groundwater to contact the WPs (proposed 10 CFR 63.21(c)(18) [64 FR 8640]).

Discuss sealing equipment and procedures for shafts. Provide a figure that shows the shaft seal operations sequence diagram.

Discuss sealing equipment and procedures for ramps. Provide a figure that shows the ramp seal operations sequence diagram.

6.3.2 Plugs and Bulkheads

Describe the purpose and scope of the plugs and bulkheads used in the EBS design.

Provide the required level of detail, as outlined in Section 6.1.2, for the plugs and bulkheads portion of the EBS and as supplemented by the information below.

Discuss plugs and bulkheads not considered part of the sealing system.. Provide a figure that shows the plug and bulkhead operations sequence diagram.

6.3.3 Backfill

Describe the purpose and scope of the backfill used in the EBS design.

Provide the required level of detail, as outlined in Section 6.1.2, for the backfill portion of the EBS and as supplemented by the information below.

Include discussion of the backfill materials, installation equipment, and planned installation operation. Describe aspects of the backfill design that will help keep radioactive material from reaching the surface of the repository, and help keep groundwater from contacting the WPs (proposed 10 CFR 63.21(c)(18) [64 FR 8640]).

Describe how the effects of backfill and the timing of its emplacement have been considered in the repository thermal loading analysis [CLST 6.1] (NRC 1998b). If the ceramic coating option is chosen for the WPs, discuss compatibility of the ceramic coating material with the backfill material. Discuss the effect of the combination of ceramic coating and backfill on WP lifetime [CLST 6.3] (NRC 1998b). Discuss the role of backfill in repository performance. Ensure that the repository performance function of backfill as discussed in this chapter is consistent with the EBS PA in Section 8.6. Describe and justify the use of the environmental conditions for any backfill test results not specifically collected for the repository site [CLST 6.5] (NRC 1998b).

Discuss backfill equipment for shafts. Provide a figure that shows the shaft backfill operations sequence diagram.

Discuss backfill equipment for ramps. Provide a figure that shows the ramp backfill operations sequence diagram.

6.3.4 Drip Shields

State the purpose of the drip shields, and describe their design. Provide figures as needed to show the drip shield installation sequence and illustrate the installed configuration. Provide the required level of detail, as outlined in Section 6.1.2, for the drip shields portion of the EBS, and as supplemented by the information below.

Include discussion of the drip shield materials, installation equipment, and planned installation operation. Describe aspects of the drip shield design that will help keep radioactive material from reaching the surface of the repository, and help keep groundwater from contacting the WPs (proposed 10 CFR 63.21(c)(18) [64 FR 8640]).

Discuss the effect of drip shields combined with backfill on WP lifetime. Include the following topics in the discussion:

- Protection against rockfall provided to the WPs
- Extension of the humid air corrosion regime
- Environmental effects
- Drip shield breakdown, with subsequent mechanical impact on the WP
- Potential for crevice corrosion at the junction between the drip shield and the WP
- Potential condensate formation on the underside of the drip shield, with subsequent dripping onto the WPs [CLST 6.4] (NRC 1998b).

Describe and justify the use of the environmental conditions for any drip shield test results not specifically collected for the repository site [CLST 6.5] (NRC 1998b).

6.4 DESIGN EVALUATION

The purpose of this section is to describe the evaluation of the compliance of the EBS and its components with the appropriate design criteria. Describe the interface between the design evaluation in this section and the repository preclosure radiological safety assessment and postclosure PA described in Chapters 7 and 8, respectively. Particularly emphasize how the EBS design supports meeting the performance objectives of proposed 10 CFR 63.111 and 10 CFR 63.113 (64 FR 8640). Coordinate this discussion with the authors of Chapters 5, 7, and 8 to minimize repetition. Provide a reference to Sections 12.3.3 and 12.4.2 for a description of the EBS preclosure (performance confirmation) program.

Describe the design features of the engineered barriers that, when combined with the natural barriers of the repository, will demonstrate the ability of the repository to meet the postclosure performance objective of proposed 10 CFR 63.113(b) (64 FR 8640). The objective is for the combination of engineered and natural barriers to limit the expected annual dose to the average member of the critical group to not more than 0.25 mSv (25 mrem) TEDE annually during the first 10,000 years after permanent closure. Discuss how the EBS design contributes to meeting this performance objective. Refer to Chapter 8 for the total system performance analysis that will demonstrate compliance with this objective (proposed 10 CFR 63.113(b) [64 FR 8640].

Provide a cross-reference to Section 8.1.2 and state that it includes an assessment of the degree to which features, events, and processes are expected to materially affect compliance with proposed 10 CFR 63.113(b) (64 FR 8640), and whether these effects would be beneficial or adverse to performance of the geologic repository. State that the assessment includes the technical basis for either inclusion or exclusion of specific features, events, and processes of the geologic setting in the PA and inclusion or exclusion of degradation, deterioration, or alteration processes of engineered barriers in the PA, including those processes that would adversely affect the performance of natural barriers (proposed 10 CFR 63.21(c)(5) [64 FR 8640]).

State that Section 8.4 provides the results of the postclosure PA for the potential repository, as required by proposed 10 CFR 63.113(c) (64 FR 8640) (Interim Guidance Section 21(c)(7) [Dyer and Horton 1999]).

State that Chapter 8 contains the description and explanation of the measures employed to support the models used in the postclosure PA (proposed 10 CFR 63.21(c)(9) [64 FR 8640]).

State that Chapter 8 contains the description of the use of expert elicitation to support the postclosure PA (proposed 10 CFR 63.21(c)(10) [64 FR 8640]).

Describe in detail the specific barriers identified in Section 8.6 as important to waste isolation (proposed 10 CFR 63.114(h) [64 FR 8640]). Provide a cross-reference to Section 8.6 for a description of the capability of the barriers, identified as important to waste isolation, to isolate waste (proposed 10 CFR 63.114(i) [64 FR 8640]). Refer to Section 8.6 for the technical basis for the description of the capability of the barriers to isolate waste (proposed 10 CFR 63.114(j) [64 FR 8640]).

Show that the analytical and numerical models used in the EBS thermal-mechanical analyses are appropriately verified, validated, and calibrated. If there are aspects of the models for which long-term experimental data are needed, continued verification and validation during performance confirmation may be alternatively addressed. In this case, discuss detailed plans and procedures for such continued activities [RDTME 3.1.7] (NRC 1998a).

Describe, explain, and justify the principles formulating the thermal-mechanical analytical methods, underlying assumptions, resulting limitations, and steps of the EBS design procedures [RDTME 3.1.9] (NRC 1998a).

Describe how the thermal-mechanical analytical methodology considers plausible, potentially important, thermal-mechanical processes appropriate to the EBS design and site characteristics [RDTME 3.1.10] (NRC 1998a).

Describe the time sequences of thermal loading used in the EBS thermal design and analyses [RDTME 3.1.12] (NRC 1998a).

Describe how changes in thermal and mechanical properties due to rock mass degradation, caused by sustained thermal-mechanical loading and extended exposure to heat and moisture, are considered in the EBS thermal-mechanical design [RDTME 3.1.13] (NRC 1998a).

Describe how roof supports, such as bolts, shotcrete, concrete, and steel liners, interaction between the rock and the roof supports, and degradation of roof supports with time under high temperature and moisture conditions are considered in the EBS thermal-mechanical analyses [RDTME 3.1.15] (NRC 1998a). Refer to Section 6.2.3.2 for the description of the EBS ground support design.

Describe how the EBS designs will support the function of the WPs, the performance of the underground facility, or the natural barriers of the geologic setting. Include discussions of solubility, oxidation/reduction reactions, corrosion, hydriding, gas generation, thermal effects, mechanical strength, mechanical stress, radiolysis, radiation damage, radionuclide retardation, leaching, fire and explosion hazards, thermal loads, and synergistic interactions. Discuss impacts of explosive, pyrophoric, and chemically reactive materials, and free liquids.

6.4.1 Engineered Barrier System Materials Performance

State the purpose of this section, which is to discuss the performance of the EBS materials described in Section 6.2.2.1, and provide the appropriate discussion.

Describe the interface between the design evaluation described in this section and the overall repository preclosure and postclosure PA described in Chapters 7 and 8.

6.4.2 Engineered Barrier System Thermal Performance

State the purpose of this section, which is to discuss the performance of the EBS thermal design described in Section 6.2.2.2, and provide the appropriate discussion.

Describe the interface between the design evaluation described in this section and the overall repository preclosure and postclosure PA described in Chapters 7 and 8.

Ensure that thermal parameter inputs used in analysis of engineered barrier performance are consistent with analogous inputs elsewhere in the LA (e.g., spent nuclear fuel heat rate).

6.4.3 Engineered Barrier System Criticality Performance

State the purpose of this section, which is to discuss the criticality performance of the EBS criticality design features described in Section 6.2.2.3, and provide the appropriate discussion.

Note that criticality analysis is primarily discussed in Chapter 5 of the LA. The authors of Chapters 5 and 6 should work together to minimize redundancy in the discussion of criticality analysis. Describe the interfaces among the design evaluations described in this section, the WP design evaluations in Chapter 5, and the overall repository preclosure and postclosure PAs described in Chapters 7 and 8.

6.4.4 Engineered Barrier System Structural Performance

State the purpose of this section, which is to discuss EBS structural performance based on the EBS structural design described in section 6.2.2.4, and provide the appropriate discussion.

Describe how the EBS seismic analysis methods are consistent with those established in *Methodology to Assess Fault Displacement and Vibratory Ground Motion Hazards at Yucca Mountain* (YMP 1997) [RDTME 3.1.11] (NRC 1998a).

Describe how the EBS thermal-mechanical design and analyses consider the effects of lithophysae for those areas of the drifts that will cross lithophysae-rich strata [RDTME 3.1.14] (NRC 1998a).

Describe the evaluation of the compliance of the EBS SSCs with the structural design criteria. Describe the interface between the design evaluation described in this section and the overall repository preclosure and postclosure PA described in Chapters 7 and 8.

Discuss the results of the analyses that demonstrate that the response of the EBS structures not important to safety or waste isolation to credible off-normal and accident conditions will not create secondary hazards for WPs or other SSCs that are important to safety or to waste isolation.

6.4.5 Engineered Barrier System Shielding Performance

State the purpose of this section, which is to discuss the performance of the EBS shielding design described in Section 6.2.2.5, and provide the appropriate discussion.

Describe the interface between the design evaluation described in this section and the overall repository preclosure and postclosure PAs described in Chapters 7 and 8.

Verify that the source terms used for EBS radiation calculations are consistent with those provided in Chapters 5 and 10.

6.5 REFERENCES

The following references were used to develop this chapter of the TGD. For the LA, this section will contain the references used to develop this chapter of the LA.

References Cited

DOE (U.S. Department of Energy) 1998. *Quality Assurance Requirements and Description.* DOE/RW-0333P, Rev. 8. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.19980601.0022.

Dyer, J.R. and Horton, D.G. 1999. "Interim Guidance Pending Issuance of New Nuclear Regulatory Commission (NRC) Regulations for Yucca Mountain, Nevada." Letter from J.R. Dyer (DOE/YMSCO) to D.R. Wilkins (CRWMS M&O), June 18, 1999, OL&RC:AVG:1435, with enclosure, "Interim Guidance Pending Issuance of New NRC Regulations for Yucca Mountain." ACC: MOL.19990712.0039.

NRC (U.S. Nuclear Regulatory Commission) 1998a. *Issue Resolution Status Report Key Technical Issue: Repository Design and Thermal-Mechanical Effects.* Rev. 1. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.19981130.0219.

NRC 1998b. *Issue Resolution Status Report Key Technical Issue: Container Life and Source Term.* Rev. 1. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.19990105.0081.

YMP (Yucca Mountain Site Characterization Project) 1996. *Preclosure Seismic Design Methodology for a Geologic Repository at Yucca Mountain.* Topical Report. YMP/TR-003-NP, Rev. 1. Las Vegas, Nevada: Yucca Mountain Site Characterization Office. ACC: MOL.19970114.0027.

YMP 1997. *Methodology to Assess Fault Displacement and Vibratory Ground Motion Hazards at Yucca Mountain.* Topical Report. YMP/TR-002-NP, Rev. 1. Las Vegas, Nevada: Yucca Mountain Site Characterization Office. ACC: MOL.19971016.0777.

Codes, Standards, and Regulations

10 CFR (Code of Federal Regulations) 20. Energy: Standards for Protection Against Radiation. Readily available.

64 FR (Federal Register) 8640. Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, Nevada. Proposed rule 10 CFR 63. Readily available.

INTENTIONALLY LEFT BLANK

CHAPTER 7. PRECLOSURE RADIOLOGICAL SAFETY ASSESSMENT

This chapter provides guidance to the authors of Chapter 7 of the License Application (LA) to describe the safety assessments for radiological exposures to the public and repository workers during design basis events (DBEs). The assessments will demonstrate that radioactive waste can be handled in the monitored geologic repository (MGR) without unreasonable risk to the public or repository worker. Authors preparing the LA shall use this document for guidance and must read the Introduction and Appendix B before writing their respective sections. By following this technical guidance document, the LA authors will produce a chapter that will enable the U.S. Nuclear Regulatory Commission (NRC) to determine that there is reasonable assurance the repository can be constructed and safely operated.

7. PURPOSE AND SUMMARY

State that the purpose of Chapter 7 is to provide the preclosure safety assessments for radiological exposures to the public and repository workers during normal operations and DBEs and to describe the location, in other chapters, of information that provides input to the safety analyses in Chapter 7. State that Chapter 2 provides an overview description of how the integrated safety analysis (ISA) is addressed in the LA. State that this overview includes cross-reference information to where the elements of the ISA are addressed in the LA (proposed 10 CFR 63.21(c)(2) [64 FR 8640]). Provide a summary of the information contained in the chapter and a discussion of the organization of the chapter.

7.1 REQUIREMENTS

Describe the preclosure radiation exposure limits that are applicable to the potential repository. Discuss the inclusion, by reference, of the requirements in 10 CFR 20. The requirements of interest in this chapter pertain to preclosure radiological safety. As called out in 10 CFR 20 and the proposed 10 CFR 63.111(a) (64 FR 8640), the potential exposures to members of the public and repository workers should be estimated for all DBEs. As described in proposed 10 CFR 63.2 (64 FR 8640), the NRC defines DBE Categories 1 and 2 as encompassing the full spectrum of events, including normal operations, anticipated operational occurrences, and accidents.

Describe the scope of Chapter 7, which will demonstrate that the repository will comply with all applicable radiological requirements for the preclosure period as defined in the proposed 10 CFR 63.111(a) (64 FR 8640) as required by proposed 10 CFR 63.111(c) (64 FR 8640). This evaluation of compliance will consist of DBE identification, categorization, and evaluation against the defined applicable radiological limits. Information that is design-specific and site-specific will be used in the evaluations. Describe how Chapter 7 will provide portions of the ISA (proposed 10 CFR 63.21(c)(2) [64 FR 8640]), such as that portion required by Interim Guidance Section 112(b) (Dyer and Horton 1999) and proposed 10 CFR 63.112(d) (64 FR 8640). Indicate that some aspects of repository performance considered to have an impact on radiological safety will be addressed in other chapters. In particular, the as low as is reasonably achievable philosophy, assessment of occupational doses, and organization of the radiation protection program will be addressed in Chapter 10. Also, the demonstration of compliance with the criticality requirement during preclosure operations will be documented in Section 5.3.1.

7.2 APPROACH

Describe the approach used for assessing the radiological safety of the potential repository until permanent closure. The approach will be based on the DBE approach outlined in the proposed 10 CFR 63.111(a) (64 FR 8640) for items important to radiological controls and the mandated performance of the geologic repository operations area (GROA) until permanent closure. The approach will support applicable criteria of the ISA required by the proposed 10 CFR 63.112 (64 FR 8640).

Organize the Section 7.2 overview discussion to address the following six steps:

1. Identify the conditions and sequences of events that could lead to radiation exposures, radiation levels, and releases of radioactive material to members of the public. Describe the role of the hazards analysis to systematically identify potential DBE scenarios.

2. Discuss the frequency (annual probabilities) of occurrence of these repository conditions and sequences of events and determine the regulatory limits appropriate for these conditions and sequences of events (i.e., to classify them as Category 1 or Category 2 DBEs). Events having a probability of occurrence of less than one chance in 10,000 of occurring before permanent closure of the repository are treated as beyond DBEs.

3. Analyze source terms (quantities, concentrations, and specifications of potential releases and direct radiation exposure levels) that are expected to occur for applicable Category 1 and Category 2 DBEs.

4. Identify and analyze receptors (locations and lifestyles of individuals) for potential exposures and radiation levels, and releases of radioactive material as appropriate, respectively, for Category 1 and Category 2 DBEs.

5. Describe the models used to determine potential radiological impacts within the exposed population as appropriate for Category 1 and Category 2 DBEs. Radiological impacts of all Category 1 events (including normal operations and anticipated operational occurrences) are to be considered to determine the average annual exposures and releases to demonstrate compliance with the annual public dose limit of 0.25 mSv (25 mrem) (Interim Guidance Section 111(a)(2) [Dyer and Horton 1999]). Radiological impacts for Category 2 events are analyzed as consequences of each event like an independent occurrence.

6. Describe with appropriate technical basis that for normal operating events, and assuming the occurrence of Category 1 DBEs, the performance objectives of the proposed 10 CFR 63.111(a)(1) (64 FR 8640) are met for protection against radiation exposures and release of radioactive material as required by 10 CFR 20. Discuss how the consequences of Category 1 DBEs, combined with the expected radiological exposures from normal operations, are within the regulatory limits specified in 10 CFR 20 (proposed 10 CFR 63.111(a)(1) [64 FR 8640]).

For each step above, identify sections of this chapter in which detailed information is provided. For topics outside the scope of this chapter, such as definition of source terms, identify the other chapters where these topics are addressed.

Ensure that the approach discussion describes the following:

- The extent to which a probabilistic risk assessment approach is applied in assessing the performance of the repository, as endorsed by the NRC in policy statements (60 FR 42622; NRC 1995) and Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis* (NRC 1988).

- The extent to which a purely deterministic approach, as established in previous licensing of nuclear facilities, is used in assessing the performance of the potential repository and the rationale for such augmentation. For example, offsite dose analyses are expected to use deterministic approaches in accordance with regulatory precedents for radionuclide release fractions and assumed fractions of waste form damage.

- The extent to which the approach used parallels from the nuclear industry in developing the Safety Analysis Report and the LA for dry storage facilities or dry cask systems, insofar as that approach has been previously accepted by NRC. Relate the discussion to the pertinent review plan elements in Chapter 9, *Radiation Protection Evaluation*, and Chapter 12, *Accident Analyses*, of NUREG-1567 (NRC 1996a) and in NUREG-1536 (NRC 1997), as applicable. Provide a discussion that during preclosure operation, there exist administrative limits, procedures, training, and corrective and preventive maintenance programs that ensure the equipment and personnel will be available to respond to DBEs appropriately.

- Elements of conservatism applied in the approach. For example, analyses of probabilities, exposures, and doses may be based on best-estimate parameters for Category 1 DBEs and conservative or bounding parameters for Category 2 DBEs.

- The process by which potential accident scenarios were systematically identified, screened, and categorized. The discussion should provide tabulation of all internal and external events initially considered and of the Category 1, Category 2, and beyond DBEs.

7.2.1 Event Type

Describe the method used to identify and systematically analyze naturally occurring and human-induced hazards at the GROA, including a comprehensive identification of potential DBEs for the repository (Interim Guidance Section 112(b) [Dyer and Horton 1999]). Describe the preliminary hazard analysis process for initial identification and qualitative screening of candidate conditions and events. Describe how data pertaining to the Yucca Mountain site and the surrounding region are used, to the extent necessary, for identifying naturally occurring and human-induced hazards at the repository operations area (proposed 10 CFR 63.112(c) [64 FR 8640]). Discuss the technical basis for either inclusion or exclusion of specific, naturally

occurring, and human-induced hazards in the analysis (proposed 10 CFR 63.112(d) [64 FR 8640]).

As appropriate, document the screening derived from NUREG-0800 (NRC 1987), Regulatory Guide 1.91, Rev. 1 (NRC 1978), or similar regulatory precedents.

Describe the two categories of DBEs as defined in the proposed 10 CFR 63.2 (64 FR 8640). Category 1 is the set of DBEs expected to occur one or more times before permanent closure of the GROA. Category 2 is the set of DBEs with at least one chance in 10,000 of occurring before permanent closure of the geologic repository. Cite NRC interpretations (per NRC 1996b) that DBEs comprise a sequence of events. Describe how events having less than one chance in 10,000 of occurring before permanent closure of the geologic repository are treated as beyond DBEs and will be used to demonstrate margins to compliance.

Describe how DBEs, other than "normal operations" DBEs, can be initiated by internal events, such as mechanical or other failures; human error; or external events, both human-initiated (e.g., aircraft crashes) and natural phenomena (e.g., earthquakes and tornadoes). Describe how event-tree modeling is applied to define potential accident sequences (or scenarios) that could result in release of radionuclides to the environment. An accident sequence is thus described as an initiating event followed by one or more events that propagate, or fail to mitigate, the accident sequence so that release of radionuclides can occur. Note that the frequency for such sequences is the product of the frequency of the initiating event multiplied by the conditional probability of occurrence of each subsequent event in the sequence.

Describe how numerous DBEs within a given category may lead to similar, if not identical, consequences. Note that to avoid detailed evaluation of many similar events, the events will be grouped into subsets of events that lead to comparable waste form and container damage and thus to similar radionuclide releases. Typically, within each subset, only the bounding release case is considered in detailed analysis.

A discussion of incredible events (those with a probability below the lower limit for Category 2) shall be included for consideration of radiological consequences. This discussion is intended to demonstrate the limited hazard potential of operations being performed and the substantial defense in depth inherent in the design.

7.2.2 Radionuclide Releases

For the bounding event in each subset of DBEs, provide estimates of the types and quantities of radioactive releases that are calculated. Provide sufficient information to characterize the releases for each waste form. For each waste form, present comprehensive tables of the amount of material at risk in each unit, the total number and annual rate of receipt and emplacement, and the radioisotopic inventory, as appropriate. State that Section 5.1 of the LA also provides similar information. Describe the bases for estimating the amount of radioactivity released at the location of a DBE. Describe the bases for quantifying the fractions of gaseous, volatile, and particulate forms of radionuclides for each waste form.

Although release fraction bases may be documented in project reports, note the extent to which licensing precedents have been incorporated into such bases, where appropriate, or note that exception has been taken to such precedents. Summarize the bases and assumptions used to establish the fractions of available inventories of each waste form released to the environment in various kinds of DBEs (proposed 10 CFR 63.21(c)(12)[64 FR 8640]).

7.2.3 Radionuclide Transport and Radiation Dose

Discuss the radionuclide pathways, computational methodologies, and dose conversion factors used to arrive at dose predictions for workers and the public. Identify releases and dose calculations for Category 1 DBEs. Describe the process for combining the radiological consequences or releases from normal operations presented in Chapter 10 with the results of the Category 1 DBE releases to demonstrate compliance with the annual public dose limit of 0.25mSv (25 mrem) (Interim Guidance Section 111(a)(2) [Dyer and Horton 1999]).

Describe the method used to calculate radionuclide transport and the resultant radiological consequences to appropriate receptors. This discussion must cover the transport of radionuclides within the facility (i.e., the leak-path factors for the assembly transfer cells, confinement barriers, and filter systems) and outside the facility (atmospheric dispersion) to and beyond the boundaries of the facility. Discuss the techniques used for calculating radiological consequences for offsite exposures. Describe any computer software used to calculate radionuclide transport and dose consequences and provide the bases for asserting NRC acceptance of such hardware.

Compare the radiological consequences for each bounding event with the mandated limits and demonstrate compliance. Describe the process whereby doses associated with each Category 1 event will be considered in demonstrating compliance. The limits specified in 10 CFR 20 for repository workers and members of the public apply to Category 1 events. Provide bases for defining locations of affected workers. Describe the methods for calculating radiation transport and resulting doses to affected workers if different from the methods used for calculating offsite exposures. Refer to Chapter 10 for a discussion on worker exposure due to normal operations.

Summarize by describing how the approach described in Section 7.2 demonstrates compliance with the proposed repository objectives.

7.3 REPOSITORY DESIGN INPUT TO SAFETY ASSESSMENT

Most of the information for this section will be included in other chapters in the LA and can be incorporated here by reference. For the preclosure radiological assessment, describe the analysis of the performance of the surface (Section 4.1), waste package (Section 5.1), and subsurface (Section 6.1) design structures, systems, and components (SSCs), which identify those SSCs important to safety. Identify and describe the controls relied upon to limit or prevent potential DBEs or mitigate their consequences. Include salient details of the preventive and mitigative features for those systems. Examples of such systems may include heating, ventilation, and air conditioning; emergency power generation; monitoring systems; and emergency response systems. Refer to the specific design chapters by section for additional explanations of the design considerations related to DBEs and the system features to limit or prevent potential DBEs

(Interim Guidance Section 112(e) [Dyer and Horton 1999]). Describe the SSCs and their features that are relied upon to prevent the DBEs.

Begin each of the following sections by stating the purpose of the information as it relates to the overall purpose of this chapter.

7.3.1 Geologic Repository Operations Area Description

7.3.1.1 Surface Facilities

Describe and justify assumptions for inputs to the safety assessment of salient features of surface facilities used for radiological safety, such as heating, ventilation, and air conditioning; emergency power generation; monitoring systems; and emergency shutdown systems as appropriate. Describe the SSCs and their features relied upon to mitigate DBEs. Include identifications and descriptions of controls relied upon to limit or prevent potential DBEs or mitigate their consequences. Refer to other chapters for information regarding the process used to classify SSCs important to safety (Chapter 2) and detailed description of the features relied upon for prevention and mitigation of consequences (Sections 4.1, 5.2, and 9.1) (Interim Guidance Section 112(e) [Dyer and Horton 1999]).

7.3.1.2 Subsurface Facilities

Describe and justify assumptions for inputs to the safety assessment of salient features of subsurface facilities used for radiological safety. Describe the SSCs and their features relied upon to mitigate DBEs. Include identifications and descriptions of controls relied upon to limit or prevent potential DBEs or mitigate their consequences. Refer to other chapters for information regarding the process used to classify SSCs important to safety (Chapter 2) and detailed description of the features relied upon for prevention and mitigation of consequences (Sections 4.1, 5.2, and 6.1) (Interim Guidance Section 112(e) [Dyer and Horton 1999]).

7.3.2 Waste Package Description

Describe and justify assumptions for input into the safety assessment of salient features of the waste package with respect to radiological safety. Incorporate details by reference to Section 5.2. Discuss only information relevant to the integrity of the container. As appropriate, describe the relevant differences among waste packages used for the variety of waste forms. Include identifications and descriptions of controls relied upon to limit or prevent potential DBEs or mitigate their consequences (Interim Guidance Section 112(e) [Dyer and Horton 1999]).

7.4 SOURCE TERM DEVELOPMENT

Provide a description of the amount and radiological parameters of the waste forms selected to represent the bounding source terms for DBE consequence analyses. The amounts (material-at-risk) of each type of waste form should be listed for each location in the surface and subsurface for which DBEs have been identified. Provide the radioisotope inventory for each waste form used in DBE consequence analyses. If the inventory has been truncated to use only certain isotopes in the consequence analyses, provide the bases for not including a given isotope (e.g., it

is a negligible contributor to total consequences). As appropriate, describe the bases for the assumption of the amount of material-at-risk (e.g., residence times and throughput for each waste handling process station). Reference Section 5.1 for additional information, as appropriate. Describe how this section, along with other chapters and sections of the LA, demonstrates compliance with proposed 10 CFR 63.21(c)(12) (64 FR 8640).

Begin each of the following sections by stating the purpose of the information as it relates to the overall purpose of this chapter.

7.4.1 Light Water Reactor Spent Nuclear Fuel

Describe and justify assumptions of the form, quantities, and physical characteristics of the spent nuclear fuel (SNF) that are inputs to the safety assessment. Describe the estimated radionuclide inventory to the extent that it affects release, transport, or dose. Make appropriate reference to the source term information in Section 9.6 and the waste description in Section 5.1. Include a discussion of the characteristics of gaseous elements, volatile elements, and elements in solid form (refractory), including activated corrosion or wear products from reactor systems and the applicable radiation shielding source term data (i.e., emission rates and energy spectra). Describe how SNF from various types of light water reactors is grouped according to similar physical characteristics and how bounding source terms are developed for each group (proposed 10 CFR 63.21(c)(12) [64 FR 8640]).

7.4.2 High-Level Radioactive Waste

Describe and justify assumptions of the form, quantities, and physical characteristics of the waste as input to the safety assessment. Describe the estimated radionuclide inventory to the extent that it affects release, transport, or dose. Make appropriate reference to the source term information in Section 9.6 and the waste description in Section 5.1. Include a discussion of characteristics of the glass, volatile elements, and elements in solid form, and the types, energies, and flux rates of radiation emitted. Describe how high-level radioactive waste from various sources is grouped according to similar physical characteristics and how bounding source terms are developed for each group (proposed 10 CFR 63.21(c)(12) [64 FR 8640]).

7.4.3 Radioactive Effluents and Wastes

Describe and justify assumptions of the form, quantities, and physical characteristics of the effluents and low-level waste generated at the repository as input to the safety assessment. Describe the estimated radionuclide inventory that results in release, transport, or dose. Make appropriate reference to the source term information in Section 9.6. Demonstrate that, for normal operating events and assuming the occurrence of Category 1 DBEs, the performance objectives of proposed 10 CFR 63.111(a)(1) (64 FR 8640) are met for protection against radiation exposures and release of radioactive material as required by 10 CFR 20. Discuss how the consequences of Category 1 DBEs combined with the expected radiological exposures from normal operations are within the regulatory limits specified in 10 CFR 20.

7.4.4 Other U.S. Department of Energy Spent Nuclear Fuel

Discuss the salient features of the other U.S. Department of Energy (DOE) SNF waste forms, including a description of these waste forms as input to the safety assessment. Refer to the waste description in Section 5.1. Describe how the DOE SNF from various sources is grouped according to similar physical characteristics and how bounding source terms are developed for each group (proposed 10 CFR 63.21(c)(12) [64 FR 8640]).

7.4.5 Plutonium Contained in U.S. Department of Energy High-Level Radioactive Waste

Describe and justify assumptions of the salient features of the DOE plutonium waste as input to the safety assessment. Provide a summary of the kind, amount, source, and specification of the waste. Describe the estimated radionuclide inventory to the extent that it affects release, transport, or dose (proposed 10 CFR 63.21(c)(12) [64 FR 8640]).

7.5 DESIGN BASIS EVENT DEFINITION

Begin each of the following sections by stating the purpose of the information as it relates to the overall purpose of this chapter.

7.5.1 Design Basis Event Categorization

Present a discussion of the DBEs considered and general descriptions of the processes used to screen and select the set of candidate events for further consideration and evaluation. Describe the systematic review of all events and the data used to classify them as Category 1 or 2. Describe the process for performing a hazards analysis for external and internal initiating events. Summarize the results of the hazards analyses with respect to events that have been screened out or screened in as potential DBE initiators. Summarize the process for employing event-tree/fault-tree methodology and the extent to which it is employed in DBE analyses. Include the rationale for excluding any candidate event or event sequence from the set included as DBEs (proposed 10 CFR 63.112(d) [64 FR 8640]).

7.5.2 Category 1 Design Basis Events

Refer to the definition of Category 1 DBEs in proposed 10 CFR 63.2 (64 FR 8640). Define the numerical value for "the time to permanent closure" (e.g., 100 years) for the entire MGR or portions of the MGR, as appropriate.

Identify the Category 1 DBEs and the SSCs affected by the events. Describe the initiating event, event sequence, and the associated consequences. Refer to Sections 4.1, 5.1, and 6.1 where SSCs credited with preventing or mitigating the consequences of Category 1 DBEs are discussed. Provide a summary of the Category 1 DBE analysis for doses to individuals and release of radioactive material. Reference Chapter 10 for the limits established by proposed 10 CFR 63.111(a), as required by proposed 10 CFR 63.111(b)(1) (64 FR 8640). To aid discussion, segregate the "normal operations" events from the other Category 1 DBEs (i.e., anticipated off-normal events). For the latter, describe how those DBEs of a common nature are

placed into groups and how, for each group, a bounding event of consequence is identified (proposed 10 CFR 63.111(b)(1) [64 FR 8640]).

7.5.3 Category 2 Design Basis Events

Refer to the definition of Category 2 DBEs in proposed 10 CFR 63.2 (64 FR 8640). Define the numerical value for "the time to permanent closure" (e.g., 100 years) for the entire MGR or portions of the MGR, as appropriate.

Identify the Category 2 DBEs and the SSCs affected by the events. Describe the initiating events, events sequence, and the associated consequences. Provide a summary of the Category 2 DBE analysis for doses to individuals and release of radioactive material. Refer to Sections 4.1, 5.2, and 6.1 where SSCs credited with preventing or mitigating the consequences of Category 2 DBEs are discussed.

State that no individual located on or beyond any point on the boundary of the preclosure controlled area will receive either the more limiting total effective dose equivalent (TEDE) of 0.05 Sv (5 rem) or the sum of the deep dose equivalent and committed dose of 0.50 Sv (50 rem) to any individual organ or tissue (other than the lens of the eye). The lens dose equivalent shall not exceed 0.15 Sv (15 rem), and the shallow dose equivalent to skin shall not exceed 0.50 Sv (50 rem) (Interim Guidance Section 111 (b)(2) [Dyer and Horton 1999]).

7.6 CONSEQUENCE ANALYSES OF DESIGN BASIS EVENTS

Begin each of the following sections by stating the purpose of the information as it relates to the overall purpose of this chapter.

7.6.1 Release of Radioactivity in Design Basis Events

Describe the bases for estimating the amount of radioactivity released at the location of a DBE. Describe the bases for quantifying the fractions of gaseous, volatile, and particulate forms of radionuclides for each waste form. Discuss how the release fractions are applied to the source terms defined in Section 7.4 to define the quantities of radioisotopes released from each DBE listed in Section 7.5. Apply the release fractions to the source terms defined in Section 7.4 to define the quantities of radioisotopes released from each DBE listed in Section 7.5 (Interim Guidance 63.111(a)(2) ([Dyer and Horton 1999]).

7.6.2 Repository Radionuclide Transport Mechanisms

Describe the bases for calculating the leak-path factor. Describe the bases and physical mechanisms by which the radioactive releases, described in Section 7.6.1, are transported through and out of the surface or subsurface facility. In the source term, include particulate matter, vapor phases, and fission product gases. For particulate transport, include a discussion of mitigating features, events, and processes (i.e., fallout, plateout, and washout) in the various consequence evaluations. Include mitigating effects of the ventilation system and the operation of the radiological monitoring system (Interim Guidance Section 111(a)(2) [Dyer and Horton 1999]).

7.7 ATMOSPHERIC DISPERSION AND DOSE CALCULATIONS

Describe the analysis of transportation of radionuclides from the point of release to the boundary of the preclosure controlled area and distances beyond, as appropriate, to assess offsite dose consequences for preclosure Category 1 and Category 2 DBEs. Describe how transportation of radionuclides to the general population, and the resultant radiological consequences, are relevant to the purpose of this chapter. Describe how doses to the public are estimated using standard codes, methods, and established dose conversion factors.

Describe the analysis of occupational dose to repository workers for Category 1 DBEs other than the "normal operations" DBE (Interim Guidance Section 111(a)(2) [Dyer and Horton 1999]). Begin each of the following sections by stating the purpose of the information as it relates to the overall purpose of the chapter.

The following regulatory guides provide applicable information for methods and assumptions used for calculating radionuclide dispersions and radiological consequences:

- Regulatory Guide 1.3, *Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors* (AEC 1974a)
- Regulatory Guide 1.4, *Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors* (AEC 1974b)
- Regulatory Guide 1.25, *Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors* (AEC 1972)
- Regulatory Guide 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants* (NRC 1982)
- Regulatory Guide 1.111, *Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors* (NRC 1977a)
- Regulatory Guide 1.112, *Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors* (NRC 1976)
- Regulatory Guide 1.113, *Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I* (NRC 1977c)
- Regulatory Guide 1.109, *Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I* (NRC 1977b)
- Regulatory Guide 4.20, *Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees other than Power Reactors* (NRC 1996b).

7.7.1 Meteorological Data and Radionuclide Dispersion

Describe site-specific and other relevant meteorological data, providing a reference to Chapter 3, as appropriate. Describe how the data are used to generate the values of atmospheric dispersion coefficients, which are inputs to the safety assessment. Any special atmospheric dispersion conditions applicable to the site and having radiological repercussions shall be identified and assessed for impact on dose predictions. Describe site-specific and other relevant meteorological data, providing a reference to Chapter 3, as appropriate. Regulatory Guide 1.111 (NRC 1977a) and Regulatory Guide 1.145 (NRC 1982) provide useful and applicable guidance on the methods for calculating radionuclide dispersion (proposed 10 CFR 63.112(c) [64 FR 8460]).

7.7.2 Radiological Dose Estimates

Discuss radiological consequences for the events considered. Discuss the process of calculating doses to people after radionuclide dispersion has been calculated. This process includes selection of the exposure scenario in accordance with regulatory guidance, selection of the important pathways, modeling of environmental transport of radionuclides, and determination of radionuclide intakes and resultant doses.

For Category 1 DBEs, describe how the annual doses to workers and members of the public are calculated. Regulatory Guide 1.109 (NRC 1977b) may provide useful and applicable guidance on the methods for calculating doses to the "maximum exposed individual" for Category 1 events. Discuss use of "best estimate" parameters in the calculation of Category 1 doses, as appropriate. For Category 1 events, discuss the as low as is reasonably achievable issues, as appropriate.

For Category 2 DBEs, only the doses to members of the public on or beyond any point on the boundary of the preclosure controlled area are calculated. Regulatory Guide 1.3 (AEC 1974a), Regulatory Guide 1.4 (AEC 1974b), and Regulatory Guide 1.25 (AEC 1972) may provide useful and applicable guidance on the methods for calculating doses to the public for Category 2 events. Discuss use of "conservative" or "bounding" parameters for the calculation of Category 2 doses, as appropriate.

For each evaluation, identify the exposure scenarios that result in the appropriate bounding dose to be compared with the limits defined for the proposed repository.

Demonstrate that multiple pathways have been considered in the dose estimates. Include a discussion of TEDE and a discussion of committed organ doses from inhalation and ingestion. Regulatory Guide 8.34 (NRC 1992) may provide useful and applicable guidance on the methods for calculating occupational doses.

7.8 DEMONSTRATION OF COMPLIANCE

Summarize how Chapter 7 of the LA demonstrates compliance, either wholly or in conjunction with other chapters and sections, with applicable sections of proposed 10 CFR 63.21(c)(2), proposed 10 CFR 63.21(c)(12), proposed 10 CFR 63.111(a)(1), proposed 10 CFR 63.111(b)(1), proposed 10 CFR 63.112(b), proposed 10 CFR 63.112(c), proposed 10 CFR 63.112(d), and

proposed 10 CFR 63.113 (64 FR 8640); and Interim Guidance Section 111(a)(2), Interim Guidance Section 111(b)(2), and Interim Guidance Section 112(e) (Dyer and Horton 1999).

7.8.1 Category 1 Design Basis Events-Demonstration of Compliance with Proposed Regulation and Interim Guidance

Demonstrate compliance with proposed 10 CFR 63.111(a)(1) (64 FR 8640) and Interim Guidance Section 111(a)(2) (Dyer and Horton 1999). Describe the radiological consequences of Category 1 DBEs that occur during operation of the repository until permanent closure. Apply the methodology described in Section 7.2.3 to consider all Category 1 events, including anticipated operational occurrences and normal operations. Provide a summary table showing the calculated dose for each off-normal event and the estimated annual probability of the event to demonstrate compliance with the annual public dose limit of 0.25 mSv (25 mrem). Compare the radiological consequences with the applicable limits of, and demonstrate compliance to, the proposed 10 CFR 63.111(a)(1) (64 FR 8640) and Interim Guidance Section 111(a)(2) (Dyer and Horton 1999). In particular, demonstrate compliance with the limits of 10 CFR 20. Describe instances where mitigating features or administrative limits are required to show protection for particular DBEs. Conclude that the requirements of proposed 10 CFR 63.111(a)(1) (64 FR 8640) and Interim Guidance Section 111(a)(2) (Dyer and Horton 1999) will be met, assuming occurrence of Category 1 DBEs.

7.8.2 Category 2 Design Basis Events-Demonstration of Compliance with Interim Guidance

Demonstrate compliance with Interim Guidance Section 111(b)(2) (Dyer and Horton 1999). Describe the radiological consequences of Category 2 DBEs that occur during operation of the repository until permanent closure. Compare the predicted radiological consequences with applicable limits. Describe instances where mitigating features or administrative limits are required to show protection for particular DBEs. Provide a statement that no individual located on or beyond any point on the boundary of the preclosure controlled area will receive either the more limiting TEDE of 0.05 Sv (5 rem) or the sum of the deep dose equivalent and the committed dose equivalent of 0.50 Sv (50 rem) annually to any individual organ or tissue (other than the lens of the eye). Also state that the annual lens dose equivalent shall not exceed 0.15 Sv (15 rem) and the shallow dose equivalent to skin shall not exceed 0.50 Sv (50 rem). Conclude that the requirements of Interim Guidance Section 111(b)(2) (Dyer and Horton 1999) will be met, assuming occurrence of Category 2 DBEs.

7.9 REFERENCES

The following references were used to develop this chapter of the technical guidance document. For the LA, this section will contain the references used to develop this chapter of the LA.

References Cited

AEC (U.S. Atomic Energy Commission) 1972. *Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors*. Regulatory Guide 1.25, Rev. 0. Washington, D.C.: U.S. Atomic Energy Commission. Readily available.

AEC 1974a. *Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors*. Regulatory Guide 1.3, Rev. 2. Washington, D.C.: U.S. Atomic Energy Commission. Readily available.

AEC 1974b. *Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors*. Regulatory Guide 1.4, Rev. 2. Washington, D.C.: U.S. Atomic Energy Commission. TIC: 4651.

Dyer, J.R. and Horton, D.G. 1999. "Interim Guidance Pending Issuance of New Nuclear Regulatory Commission (NRC) Regulations for Yucca Mountain, Nevada." Letter from J.R. Dyer (DOE/YMSCO) to D.R. Wilkins (CRWMS M&O), June 18, 1999, OL&RC:AVG:1435, with enclosure, "Interim Guidance Pending Issuance of New NRC Regulations for Yucca Mountain." ACC: MOL.19990712.0039.

NRC (U.S. Nuclear Regulatory Commission) 1976. *Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors*. Regulatory Guide 1.112, Rev. 0-R. Washington, D.C.: U.S. Nuclear Regulatory Commission. Readily available.

NRC 1977a. *Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors*. Regulatory Guide 1.111, Rev. 1. Washington, D.C.: U.S. Nuclear Regulatory Commission. Readily available.

NRC 1977b. *Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I*. Regulatory Guide 1.109, Rev. 1. Washington, D.C.: U.S. Nuclear Regulatory Commission. Readily available.

NRC 1977c. *Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I*. Regulatory Guide 1.113, Rev. 1. Washington, D.C.: U.S. Nuclear Regulatory Commission. Readily available.

NRC 1978. *Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants*. Regulatory Guide 1.91, Rev. 1. Washington, D.C.: U.S. Nuclear Regulatory Commission. Readily available.

NRC 1982. *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*. Regulatory Guide 1.145, Rev. 1. Washington, D.C.: U.S. Nuclear Regulatory Commission. Readily available.

NRC 1987. *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants.* NUREG-0800. Washington, D.C.: U.S. Nuclear Regulatory Commission. Readily available.

NRC 1988. *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis.* Regulatory Guide 1.174. Washington, D.C.: U.S. Nuclear Regulatory Commission. Readily available.

NRC 1992. *Monitoring Criteria and Methods to Calculate Occupational Radiation Doses.* Regulatory Guide 8.34. Washington, D.C.: U.S. Nuclear Regulatory Commission. Readily available.

NRC 1995. *Framework for Applying Probabilistic Risk Analysis in Reactor Regulation.* SECY-95-280. Washington, D.C.: U.S. Nuclear Regulatory Commission. TIC: 232400.

NRC 1996a. *Standard Review Plan for Spent Fuel Dry Storage Facilities.* NUREG-1567. (Draft Report for Comment). Washington, D.C.: U.S. Nuclear Regulatory Commission. TIC: 226657.

NRC 1996b. *Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees other than Power Reactors.* Regulatory Guide 4.20, Rev. 0. Washington, D.C.: U.S. Nuclear Regulatory Commission. Readily available.

NRC 1997. *Standard Review Plan for Dry Cask Storage Systems.* NUREG-1536. Washington, D.C.: U.S. Nuclear Regulatory Commission. TIC: 232373.

Codes, Standards, and Regulations

10 CFR (Code of Federal Regulations) 20. Energy: Standards for Protection Against Radiation. Readily available.

60 FR (Federal Register) 42622. Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities: Final Policy Statement. Readily available.

64 FR 8640. Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, Nevada. Proposed rule 10 CFR 63. Readily available.

CHAPTER 8. PERFORMANCE OF THE REPOSITORY AFTER PERMANENT CLOSURE

This chapter of the License Application (LA) provides a discussion of the postclosure performance assessment (PA) related information needed to support LA. The authors will produce a chapter that will describe PA methodology and will provide analyses evaluating the probable postclosure behavior of the repository system. In general, the LA that will be submitted for construction authorization must provide sufficient information to enable the U.S. Nuclear Regulatory Commission (NRC) to determine there is reasonable assurance that the types and amounts of radioactivity described in the application can be received, possessed, and disposed of in the monitored geologic repository. To support that conclusion, Chapter 8 must demonstrate that a rigorous total system performance assessment (TSPA) has been performed to support the NRC determination of reasonable assurance that, on the basis of the record before the NRC, the expected annual dose to the average member of the critical group, over the compliance period, meets the postclosure performance objective.

Authors preparing the LA shall use this document for guidance and must read the Introduction and Appendix B before writing their respective sections.

8. PURPOSE AND SUMMARY

Provide a description of the purpose of Chapter 8 of the LA. The purpose is to demonstrate, with reasonable assurance, the ability of the geologic repository to meet performance objectives for the MGR after permanent closure, as specified in proposed 10 CFR 63.113(b) (64 FR 8640). State that the chapter will describe the results of the PA of the proposed geologic repository for the period after permanent closure as required by Interim Guidance Section 21(c)(7) (Dyer and Horton 1999). Describe how an assessment of the ability of the proposed geologic repository to limit radiological exposures in the event of limited human intrusion into the engineered barrier system (EBS) has been performed as required by Interim Guidance Section 113(d) (Dyer and Horton 1999). Explain that the analysis shall meet the requirements of Interim Guidance Section 114 (Dyer and Horton 1999) and use the reference biosphere and critical group specified in Interim Guidance Section 115 (Dyer and Horton 1999).

Describe how the chapter provides information necessary and sufficient to support a determination by the NRC that there is reasonable assurance that the types and amounts of radioactive material described in this LA can be received, possessed, and disposed of in the monitored geologic repository without unreasonable risk to the health and safety of the public. State that complete assurance that the performance objectives will be met is not expected. A reasonable assurance, on the basis of the record before the NRC, that the performance objectives will be met is the standard. The TSPA described in this chapter supports the NRC finding of reasonable assurance (proposed 10 CFR 63.101(a)(2) [64 FR 8640]).

Briefly summarize the approach used to analyze the Yucca Mountain repository system and refer the reader to more detailed discussions later in the chapter. Describe how the repository system is represented in the total system performance and the components of the overall waste disposal system. Briefly describe the key aspects of the individual components identified for the TSPA and describe how they were identified. Briefly summarize the extent and limits of the PA,

including a description of the general process involved in a TSPA and the goals, approaches, and methods for a TSPA for Yucca Mountain (proposed 10 CFR 63.113(c) [64 FR 8640]).

Briefly summarize the analyses and results of the nominal scenario evaluations and present a comparison of the results against the regulatory requirements of proposed 10 CFR 63.113(b) (64 FR 8640). Briefly summarize the assessment of the performance of the proposed geologic repository for the period after permanent closure, assuming disruptive events, including basaltic igneous activity, seismic activity, and nuclear criticality. Include a summary of the results of the human intrusion evaluation.

Provide a discussion of the organization of the chapter.

8.1 OVERVIEW

Describe the purpose of this section, which is to provide a broad overview of PA. Explain that the PA scope and basic approach will be described and that a broad overview of the multiple-barrier concept will be presented.

8.1.1 Introduction

Explain that the postclosure performance objective specified in proposed 10 CFR 63.113(b) (64 FR 8640) requires a PA to quantitatively estimate the expected annual dose over the compliance period to the average member of the critical group (proposed 10 CFR 63.102(j) [64 FR 8640]). Explain that Chapter 8 will provide the results of the assessment of the performance of the proposed geologic repository for the period after permanent closure as required by Interim Guidance 63.21(c)(7) (Dyer and Horton 1999).

8.1.2 Scope of the Performance Assessment

Describe how the ability of the geologic repository to limit radiological exposures will be demonstrated through a PA that meets the requirements specified in Interim Guidance Section 114 (Dyer and Horton 1999), uses the reference biosphere and critical group specified in Interim Guidance Section 115 (Dyer and Horton 1999), and excludes the effects of human intrusion (proposed 10 CFR 63.113(c) [64 FR 8640]). Demonstrating compliance will be performed with a rigorous TSPA for the postclosure period to support the NRC determination of reasonable assurance that, on the basis of the record before the NRC, the expected annual dose to the average member of the critical group, over the compliance period, meets the postclosure performance objective. Explain that an assessment of the ability of the proposed geologic repository to limit radiological exposures in the event of limited human intrusion into the EBS will be presented separately (proposed 10 CFR 63.21(c)(8) [64 FR 8640]).

Provide a general discussion summarizing how the features, events, and processes (FEPs) considered in the PA represent a wide range of both beneficial and potentially adverse effects on performance (e.g., the beneficial effects of radionuclide sorption, the potentially adverse effects of fracture flow or a criticality event). Describe why and how FEPs that are expected to materially affect compliance with proposed 10 CFR 63.113(b) (64 FR 8640) or are expected to be potentially adverse to performance are included in the PA, while events of very low

probability of occurrence (i.e., less than one chance in 10,000 of occurring during 10,000 years) can be excluded from the analysis (proposed 10 CFR 63.114(d) [64 FR 8640]). If the U.S. Department of Energy (DOE) has chosen to analyze very low probability of occurrence events, briefly identify the events and explain why they were included.

Present an overview of the method for mathematical and numerical modeling of process and component models, including uncertainty, and the approach for combining them into an overall model and computer code. Briefly describe the use of the executive driver program (RIP V5.19.01) that links the various component codes together. Include a brief explanation of the computational work and codes that were conducted before running the actual total system computations (e.g. TOUGH2 and WAPDEG) were run.

Summarize the extent and limits of the PA.

8.1.3 Basic Approach of the Performance Assessment

Provide a brief introductory statement describing the purpose and organization of this section. State that this section provides an overview of the basic approach of the PA, including a discussion of the multiple barrier concept.

8.1.3.1 Components of the Waste Disposal System

Briefly describe the key aspects of the individual components identified for the TSPA and describe how they were identified. Briefly describe the overall repository system and the components relevant to the repository behavior. Provide a brief overview of the conceptualization of each individual process, including a description of data sources, model parameter development and computer methods used to simulate each component.

8.1.3.2 Approach to Demonstrating Compliance

Explain the basic approach to demonstrating compliance with the postclosure performance objective specified in proposed 10 CFR 63.113(b) (64 FR 8640) is to quantitatively estimate the expected annual dose over the compliance period to the average member of the critical group. Explain that a PA is a systematic analysis that identifies the FEPs (i.e., specific conditions or attributes of the geologic setting, degradation, deterioration, or alteration processes of engineered barriers) that might affect performance of the geologic repository, examines their effects on performance, and estimates the expected annual dose (proposed 10 CFR 63.102(j) [64 FR 8640]).

Describe the hierarchy of models used in PA. Discuss, at a summary level, system and subsystem models, process models, and models of potentially disruptive FEPs. Explain the relevance of information presented to overall repository performance or NRC regulations. Provide an overview of the method for mathematical and numerical modeling of each process and component described earlier and the approach for combining them into an overall model and computer code. Include a brief description of the quality assurance (QA) measures employed to establish confidence in the validity of models. Refer to the *Quality Assurance Requirements and Description* (QARD) (DOE 1998) and associated implementing procedures, as applicable.

Describe the method for mathematical and numerical modeling of process and component models, including uncertainty, and the approach for combining them into an overall model and computer code. Include a description of the information flow between the models and the computer code architecture facilitating the information flow. Describe how the component models are recoupled into one integral whole, to reassemble the analyzed pieces and pass information between them to develop reasonable assessments of overall system performance. Describe the executive driver program (RIP V5.19.01) that links the various component codes, including how component models may be coupled into RIP. Include a detailed explanation of the computational work and codes that were conducted before the actual total system computations (e.g. TOUGH2 and WAPDEG) were run. Describe the coding methods and couplings used for the major components.

8.1.3.3 Multiple Barriers

Provide an overview of the multiple-barrier concept as part of the defense-in-depth approach to support the postclosure performance objectives. Describe how it is intended that natural barriers and the EBS work in combination to enhance the resiliency of the proposed geologic repository and increase confidence that the postclosure performance objective in proposed 10 CFR 63.113(b) (64 FR 8640) will be achieved (proposed 10 CFR 63.102(h) [64 FR 8640]). Summarize the multiple-barrier analyses approach. Refer to Section 8.6 of this chapter for specific details including the results of the analysis. Describe how the results of the analyses demonstrate a robust repository with defense-in-depth and how they help ensure conformance with postclosure performance objectives (proposed 10 CFR 63.113(a) [64 FR 8640]).

8.2 SELECTION OF TOTAL SYSTEM PERFORMANCE ASSESSMENT SCENARIOS

Describe how this section of Chapter 8 provides a discussion of the scenario selection and analysis methodology for nominal and disruptive event scenarios. Explain that the human intrusion regulatory basis will also be described.

8.2.1 Scenario Selection Method

Describe the scenario selection methodology, including the identification and classification of FEPs; the screening criteria for FEPs; the scenarios constructed from retained FEPs; the screening scenarios, including the plan for identifying criteria to be used for screening scenarios to eliminate those that are logically or physically unrealistic or are expected to result in trivial consequences; and the retained scenarios.

8.2.2 Scenarios Analyzed in the Total System Performance Assessment

Describe in detail the FEPs included, the FEPs excluded and the basis for eliminating FEPs, and the approach to combining the results of scenarios. Describe the nominal and disruptive event scenarios and their probabilities. Document the rationale for the elimination or retention of any scenario considered. State that FEPs that describe the reference biosphere are consistent with present knowledge of the conditions in the region surrounding the Yucca Mountain site. Provide a reference to Chapter 3 for details regarding the conditions in the region around Yucca Mountain.

Provide the technical basis for either inclusion or exclusion of specific FEPs of the geologic setting in the PA. Demonstrate that FEPs of the geologic setting have been evaluated in detail if the magnitude and time of the resulting expected annual dose would be significantly changed by their omission (proposed 10 CFR 63.114(e) [64 FR 8640]).

Provide the technical basis for either inclusion or exclusion of degradation, deterioration, or alteration processes of engineered barriers in the PA, including those processes that would adversely affect the performance of natural barriers. Demonstrate that degradation, deterioration, or alteration processes of engineered barriers have been evaluated in detail if the magnitude and time of the resulting expected annual dose would be significantly changed by their omission (proposed 10 CFR 63.114(f) [64 FR 8640]).

Ensure that the following criteria, which deal with the effects of the waste package (WP) and EBS designs on the potential for nuclear criticality, have been addressed (NRC 1998a). As appropriate, provide reference to supporting documentation, including other LA chapters and process model reports.

- Describe credible scenarios identified during postclosure that could result in criticality. Provide a reference to the appropriate section for a demonstration that the consequences are conservatively estimated and unacceptable risks are mitigated [CLST 5.3] (NRC 1998a).
- Identify and document FEPs that may have an effect on nuclear criticality during the regulatory period and support such documentation with available data, analyses, and interpretations [CLST 5.4] (NRC 1998a).

8.2.3 Features, Events, and Processes Database

Explain that the purpose of the FEPs database is to provide a technical database for documentation of the technical basis for screening decisions. Describe what is included for each FEP entry. Describe the organization and functionality of the FEP database. Describe the QA of the database.

8.2.4 Human Intrusion

Describe the regulatory basis for human intrusion analysis and provide a broad overview of the scenario chosen to address the requirement. Describe how Interim Guidance Section 113(d) (Dyer and Horton 1999) requires that the ability of the geologic repository to continue to isolate waste from the accessible environment over the long term in the event of limited human intrusion into the EBS will be demonstrated through a separate PA that meets the requirements specified in Interim Guidance Section 114 (Dyer and Horton 1999) and uses the reference biosphere and critical group specified in Interim Guidance Section 115 (Dyer and Horton 1999). Explain that while no quantitative regulatory limit applies to the results, they will provide a qualitative indicator of the ability of the proposed geologic repository to continue to perform acceptably following human intrusion. Provide a reference to Section 8.5.1 for a more complete description of the stylized analysis of human intrusion.

8.3 SYSTEM AND SUBSYSTEM DESCRIPTIONS

Explain that the purpose of this section is to describe models, abstractions, and databases, as appropriate, and to provide supporting information, including the technical basis and supporting analyses.

8.3.1 Subsystem Models and Data

Explain how the PA used to demonstrate compliance with proposed 10 CFR 63.113(b) (64 FR 8640) includes data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the Yucca Mountain site and the surrounding region to the extent necessary. Include information on the design of the EBS used to define parameters and conceptual models used in the assessment (proposed 10 CFR 63.114(a) [64 FR 8640]). Provide references, as appropriate, to supporting documentation containing data or other pertinent information. Refer to Chapter 6 for information on the EBS.

For each subsection in Section 8.3.1, provide the information described below:

- Describe the detailed process models, submodels, abstractions, and databases used in the PAs. Provide the technical basis for models and abstractions used in the PA, such as comparisons made with outputs of detailed process-level models or empirical observations (e.g., laboratory testing, field investigations, natural analogs) (proposed 10 CFR 63.114(g) [64 FR 8640]). Include any conservative assumptions used in developing the models, and compare the results to corresponding process models or direct observations.

- Describe the measures used to support the models and abstractions. Analyses and models that will be used to assess performance of the geologic repository must be supported using an appropriate combination of such methods as field tests, in situ tests, laboratory tests that are representative of field conditions, monitoring data, and natural analog studies (proposed 10 CFR 63.21(c)(9) [64 FR 8640]).

- Explain how the PA used to demonstrate compliance with proposed 10 CFR 63.113(b) (64 FR 8640) accounts for uncertainties and variabilities in parameter values. Provide the technical basis for parameter ranges, probability distributions, or bounding values used in the PA (proposed 10 CFR 63.114(b) [64 FR 8640]). Provide a brief summary of important uncertainties that affect the conclusions supporting the intent of this chapter.

- Address probabilities and uncertainties associated with the scenarios and associated parameters to be analyzed. Discuss the techniques used to estimate probabilities, including predictive modeling and expert judgment, and describe the criteria used for each technique. Address time-dependent probabilities for transient phenomena, as appropriate. Identify alternative approaches for estimating probabilities when little theoretical, experimental, or historical data exist. Identify the factual bases and rationale for the accepted values.

- Describe the development of parameter distribution functions for the numerical models and provide the reasoning for the distributions selected. Boundary and initial conditions for the analyses must be described and justified.

- Explain how the PA used to demonstrate compliance with proposed 10 CFR 63.113(b) (64 FR 8640) considers alternative conceptual models for FEPs consistent with available data and current scientific understanding, and evaluate the effects that alternative conceptual models have on the performance of the potential geologic repository (proposed 10 CFR 63.114(c) [64 FR 8640]).

- Describe the approach to testing alternative models. Discuss the reliability, testing, and identification of alternative models for each process (e.g., unsaturated zone [UZ] flow). Where appropriate, provide reference to source material used in development of alternative model components, abstractions, etc.

- Demonstrate that the following programmatic acceptance criteria are satisfactorily addressed (NRC 1998b).

  - State that collection, documentation, and development of all data, methods, models, or computer codes included in the PA have been performed in accordance with the QARD (DOE 1998) and associated implementing procedures [TSPAI P1] (NRC 1998b), [RDTME 3.1.1] [RDTME 3.2.1] [RDTME 3.3.1] (NRC 1998c).

  - Provide an explanation of how expert assessments were used (proposed 10 CFR 63.21(c)(10) [64 FR 8640]). State that expert elicitations were conducted and documented in accordance with the requirements of the QARD (DOE 1998). Identify any expert elicitations that have been used in support of the PA, and the extent to which each was used (NRC 1996), [TSPAI P2] (NRC 1998b), [RDTME 3.1.2] [RDTME 3.2.2] [RDTME 3.3.2] (NRC 1998c).

- For process models, submodels, and abstractions used in the PAs, ensure the following technical criteria have been addressed as applicable (NRC 1998b).

  - Sufficient data (field, laboratory, and natural analog data) are available to adequately support the conceptual models, assumptions, and boundary conditions. Define all relevant parameters implemented in the TSPA [TSPAI 1] (NRC 1998b).

  - Parameter values, assumed ranges, probability distributions, and bounding assumptions used in the TSPA are technically defensible and reasonably account for uncertainties and variabilities [TSPAI 2] (NRC 1998b).

  - Alternative modeling approaches consistent with available data and current scientific understanding are investigated, and results and limitations are considered appropriately in the abstractions [TSPAI 3] (NRC 1998b).

  - Models implemented in the TSPA provide results that are consistent with output of detailed process models, empirical observations, or both [TSPAI 4] (NRC 1998b).

– TSPA adequately incorporates important design features, physical phenomena, and couplings and uses consistent and appropriate assumptions throughout the abstraction process [TSPAI 5] (NRC 1998b).

8.3.1.1 Unsaturated Zone Flow and Transport

Describe in detail the distinct components of unsaturated flow and transport, including climate, infiltration, mountain-scale flow of water, and seepage into repository emplacement drifts. Include a description of the construction of the conceptual models, implementation of the models, and results and interpretation of the analyses. Describe the important couplings with other models, major assumptions, sensitivities, and uncertainties. Ensure the requirements delineated in Section 8.3.1 have been addressed, as applicable.

Explain how the PA is based on the assumption that future climatic changes over the postclosure performance period will be consistent with the geologic record of past natural climate change in the region surrounding the Yucca Mountain site (Interim Guidance Section 114(k) [Dyer and Horton 1999]). Refer to Section 3.4 for additional details regarding the climatology of Yucca Mountain and the surrounding region. Provide reference, as appropriate, to justify the assertions being made.

Ensure that the programmatic and technical criteria delineated in Section 8.3.1, have been addressed with regard to the spatial and temporal distribution of water flow [TSPAI 2.1.1] (NRC 1998b). As appropriate, provide reference to supporting documentation including other LA chapters and process model reports.

Ensure that the programmatic and technical criteria delineated in Section 8.3.1 have been addressed with regard to the distribution of mass flux between fracture and matrix [TSPAI 2.1.2] (NRC 1998b). As appropriate, provide reference to supporting documentation, including other LA chapters and process model reports.

Ensure that the programmatic and technical criteria delineated in Section 8.3.1 have been addressed with regard to the retardation in fractures in the UZ [TSPAI 2.1.3] (NRC 1998b). As appropriate, provide reference to supporting documentation, including other LA chapters and process model reports.

Ensure that the following criteria, which address the radionuclide transport through porous rock and alluvium, have been addressed (NRC 1998d). As appropriate, provide reference to supporting documentation, including other LA chapters and process model reports.

- Demonstrate through PA calculations whether radionuclide attenuation processes such as sorption, precipitation, radioactive decay, and colloidal filtration are important to performance [RT 1.1a] [RT 2.1a] (NRC 1998d).

- For the estimation of radionuclide transport through porous rock and alluvium, either demonstrate that it has been found that radionuclide attenuation is unimportant to performance, in which case K_d is assumed to be zero and that radionuclides travel at the rate of groundwater flow, or, if radionuclide attenuation in porous rock is important to

performance, demonstrate that Criterion 2 or 3 of the Radionuclide Transport Issue Resolution Status Report has been met [RT 1.1b] [RT 2.2b] (NRC 1998d).

- For the valid application of the K_d approach, using the equation $Rf=1+(\rho K_d/n)$, demonstrate that the flow path acts as an isotropic homogeneous porous medium [RT 1.2a] [RT 2.2a] (NRC 1998d).

- For the valid application of the K_d approach, using the equation $Rf=1+(\rho K_d/n)$, demonstrate that appropriate values for the parameters, K_d, n or θ, and ρ have been adequately considered (i.e., experimentally determined or measured) [RT 1.2b] [RT 2.2b] (NRC 1998d).

- For the valid application of process models such as surface complexation, ion exchange, precipitation/dissolution, and processes involving colloidal material, demonstrate that the flow path acts as an isotropic homogeneous porous medium [RT 1.3a] [RT 2.3a] (NRC 1998d).

- For the valid application of process models such as surface complexation, ion exchange, precipitation/dissolution, and processes involving colloidal material, demonstrate that values for the parameters used in process models are appropriate [RT 1.3b] [RT 2.3b] (NRC 1998d).

- Where data are not reasonably or practicably obtained, demonstrate that expert judgement has been used and expert elicitation procedures have been adequately documented. If used, demonstrate that expert elicitations were conducted and documented in accordance with the requirements of the QARD (DOE 1998) [RT 1.4] [RT 2.4] (NRC 1998d).

- Data and models have been collected, developed, and documented under acceptable QA procedures, or if data were not collected under an established QA program, that they have been qualified under appropriate QA procedures [RT 1.5] [RT 2.5] (NRC 1998d).

Ensure that the following criterion, which addresses climate change, has been addressed (NRC 1998h). As appropriate, provide reference to supporting documentation, including other LA chapters and process model reports.

- Show that values for climatic parameters (e.g., time of onset of climate change, mean annual precipitation, mean annual temperature, etc.) used in the DOE safety case are adequately justified. Describe how appropriate scientific data were used, interpreted, and appropriately synthesized into parameters such as mean annual precipitation, mean annual temperature, and long-term climate variability. The current knowledge about these parameters, coupled with past climate change, will require that, as a bounding condition, a return to full pluvial climate (higher precipitation and lower temperatures) be considered for at least part of the 10,000-year postclosure performance period, since current information does not support the persistence of the present-day climate for a duration of 10,000 years or more. The current interpretations of paleoclimate data indicate an increase in mean annual precipitation by a factor of 2 to 3 and a decrease in

mean annual temperature of 5 to 10°C (9 to 18°F) during pluvial climate episodes [USFIC 2.4] (NRC 1998h).

- Demonstrate adequate incorporation of future climate changes and associated effects in the performance assessment. If appropriate, explain how the consequences of climate change may be coupled to other events and processes and therefore the projections of water-table rise that are used in total system performance may be different from those based solely on climate change [USFIC 2.8] (NRC 1998h).

Ensure that the following criterion, which addresses present day shallow infiltration, has been addressed (NRC 1998h). As appropriate, provide reference to supporting documentation, including other LA chapters and process model reports. Explain that estimates of the probability distribution or upper bound for present-day shallow infiltration need not be refined further if it can be demonstrated through TSPA and associated sensitivity analyses that such refinements will not significantly alter the estimate of total-system performance [USFIC 3.4] (NRC 1998h).

Ensure that the following criteria, which address matrix diffusion, have been addressed (NRC 1998h). As appropriate, reference supporting documentation, including other LA chapters and process model reports.

- Discuss whether the DOE will assume credit for matrix diffusion in the UZ. If credit is taken, show that matrix diffusion predictions are consistent with evidence for limited matrix diffusion in the UZ, including geochemical data (e.g., Yang et al. 1997) that provide evidence of geochemical disequilibrium between matrix and fracture waters in the UZ at Yucca Mountain and chlorine-36 evidence for rapid transport pathways to the repository horizon (e.g., Fabryka-Martin et al. 1996) [USFIC 6.1] (NRC 1998h).

- Discuss whether the DOE will assume that no matrix diffusion will occur in the saturated zone (SZ) (i.e., that all solutes will remain in fractures) during transport through SZ fractured rock aquifers. If credit is taken, support the DOE inclusion of matrix diffusion in SZ transport models for Yucca Mountain by both field and laboratory observations. Demonstrate that field and lab observations include tracer tests that were conducted over different distance scales and flow rates with multiple tracers of different diffusive properties. Also, show that transport model results reasonably match the results of the field tracer tests. Finally, discuss whether rock matrix and solute properties used to justify the inclusion of matrix diffusion in TSPA models fall within a range that can be supported by laboratory data [USFIC 6.2] (NRC 1998h).

8.3.1.2 Thermal Hydrology

Describe in detail thermal-hydrologic (TH) processes modeled, including the thermal-hydrologic-chemical (THC) drift-scale model, the multiscale near-field/EBS TH model, and the thermal-hydrologic-mechanical drift-scale and mountain-scale models. Include a description of the construction of the conceptual models, implementation of the models, and results and interpretation of the analyses. Describe the important couplings with other models, major assumptions, sensitivities, and uncertainties. Ensure the requirements delineated in Section 8.3.1 have been addressed, as applicable.

Ensure that the following criteria, which relate to the thermohydrologic testing program including the potential for thermal reflux to occur in the near field, have been addressed (NRC 1998e). As appropriate, provide reference to supporting documentation, including other LA chapters and process model reports.

- The DOE thermohydrologic testing program was developed under acceptable QA procedures. Data were collected and documented under purview of these procedures [TEF P1] (NRC 1998e).

- Where data are not reasonably or practicably obtained, demonstrate that expert judgement has been used and expert elicitation procedures have been adequately documented. If used, demonstrate that expert elicitations were conducted and documented in accordance with the requirements of the QARD (DOE 1998), [TEF P2] (NRC 1998e).

- Thermohydrologic tests are designed and conducted with explicit consideration of TH, thermal-chemical, and hydrologic-chemical couplings [TEF 1.1b] (NRC 1998e).

- Thermohydrologic tests are designed and conducted at different scales to discern scale effects on observed phenomena [TEF 1.1c] (NRC 1998e).

- Thermohydrologic tests are designed and conducted for temperature ranges expected for repository conditions [TEF 1.1d] (NRC 1998e).

- Thermohydrologic tests are designed and conducted to determine whether water refluxes back to the heaters during either the heating or cool-down phases of the tests [TEF 1.1e] (NRC 1998e).

- Thermohydrologic test results from other sites and programs have been tempered for application to the Yucca Mountain site [TEF 1.2] (NRC 1998e).

- If the thermohydrologic testing program is not complete at the time of LA submittal, explain why the testing program does not need to be completed for the LA [TEF 1.3] (NRC 1998e).

- TH tests are designed and conducted with explicit consideration of TH, thermal-chemical, and hydrologic-chemical couplings [TEF 1.3b] (NRC 1998e).

- TH tests are designed and conducted at different scales to discern scale effects on observed phenomena [TEF 1.3c] (NRC 1998e).

- TH tests are designed and conducted for temperature ranges expected for repository conditions [TEF 1.3d] (NRC 1998e).

- TH tests are designed and conducted to determine whether water refluxes back to the heaters during either the heating or cool-down phases of the tests [TEF 1.3e] (NRC 1998e).

Ensure that the following criteria, which deal with thermal-mechanical effects, have been addressed (NRC 1998c). As appropriate, provide reference to supporting documentation, including other LA chapters and process model reports.

- Show that the analytical and numerical models used in the thermal-mechanical analyses are appropriately verified, validated, and calibrated. If there are aspects of the models for which long-term experimental data are needed, continued verification and validation during performance confirmation may be alternatively addressed. In this case, provide a brief description of the plans and procedures for such continued activities and provide a reference to Section 12.5 for additional details on the performance confirmation program [RDTME 3.1.7] (NRC 1998c).

- Discuss how the thermal-mechanical analyses that provide the background conditions over which seismic loads are superimposed consider time-dependent jointed rock behavior [RDTME 3.2.6] (NRC 1998c).

- Discuss how the PA addresses time-dependent changes in size and shape of the emplacement drift that are due to thermally induced ground movements (rock deformations, collapse, and other changes that may affect the integrity and geometrical configuration of underground openings). Describe these changes and show that the magnitudes and distributions of the changes used are consistent with the results of thermal-mechanical analyses of the underground facility [RDTME 3.3.3] (NRC 1998c).

- Discuss how the PA addresses changes in hydrological properties (e.g., fracture porosity and permeability) due to thermally induced ground movements. Describe these changes, and ensure that the magnitudes and distributions of the changes used are consistent with the results of thermal-mechanical analyses of the underground facility [RDTME 3.3.4] (NRC 1998c).

8.3.1.3 In-Drift Geochemical Environment

Describe in detail the models developed to represent the in-drift geochemical environment, including models representing seepage and backfill interactions, precipitates and salts analysis, corrosion products, seepage and invert interactions, seepage and cement interactions, in-package chemistry, in-drift gas flux and composition, microbial communities, and in-drift colloids and concentrations. Include a description of the construction of the conceptual models, implementation of the models, major uncertainties of the models, and results and interpretation of the analyses. Describe important couplings with other models, major assumptions, sensitivities, and uncertainties. Ensure the requirements delineated in Section 8.3.1 have been addressed, as applicable.

Ensure that the programmatic and technical criteria delineated in Section 8.3.1 have been addressed with regard to the quantity and chemistry of water that contacts WPs and waste forms [TSPAI 1.1.3] (NRC 1998b). As appropriate, provide reference to supporting documentation, including other LA chapters and process model reports.

Ensure that the following criteria, which relate to the effects of coupled processes on the rate of seepage into the repository, have been addressed (NRC 1998f). As appropriate, provide reference to supporting documentation, including other LA chapters and process model reports.

- Available data relevant to both temporal and spatial variations in conditions affecting coupled THC effects on seepage and flow were considered [ENFE 1.1] (NRC 1998f).

- The DOE evaluation of coupled THC processes properly considered site characteristics in establishing initial and boundary conditions for conceptual models and simulations of coupled processes that may affect seepage and flow [ENFE 1.2] (NRC 1998f).

- Sufficient data on the characteristics of the natural system and engineered materials (e.g., the type, quantity, and reactivity of material) were collected to establish initial and boundary conditions for conceptual models and simulations of the THC processes that affect seepage and flow [ENFE 1.3] (NRC 1998f).

- Sensitivity and uncertainty analyses (including consideration of alternative conceptual models) were conducted to determine whether additional new data are needed to better define the range of values used for the input parameters [ENFE 1.4] (NRC 1998f).

- Reasonable or conservative ranges of parameters or functional relations were used to determine effects of coupled THC processes on seepage and flow. Parameter values, assumed ranges, probability distributions, and bounding assumptions are technically defensible and reasonably account for uncertainties [ENFE 1.6] (NRC 1998f).

- Uncertainty in the data due to temporal and spatial variations in conditions affecting coupled THC effects on seepage and flow was considered [ENFE 1.7] (NRC 1998f).

- The DOE evaluation of coupled THC processes properly considered the uncertainties in the characteristics of the natural system and engineered materials, such as the type, quantity, and reactivity of material, in establishing initial and boundary conditions for conceptual models and simulations of THC coupled processes that affect seepage and flow [ENFE 1.8] (NRC 1998f).

- The initial conditions, boundary conditions, and computational domain that were used in the sensitivity analyses involving coupled THC effects on seepage and flow were consistent with available data [ENFE 1.9] (NRC 1998f).

- Appropriate models, tests, and analyses that are sensitive to the THC couplings under consideration for both natural and engineering systems were used [ENFE 1.11] (NRC 1998f).

- Alternative modeling approaches (consistent with available data and current scientific understanding) were investigated, and their results and limitations were considered appropriately [ENFE 1.13] (NRC 1998f).

- A reasonable description of the mathematical models used in the analyses of coupled THC effects on seepage and flow has been provided. Include a discussion of alternative

modeling approaches (those not considered in the final analyses) and the limitations and uncertainties of the chosen model [ENFE 1.14] (NRC 1998f).

- The mathematical models used for modeling coupled THC effects on seepage and flow are consistent with conceptual models based on inferences about the near-field environment, field data and natural alterations observed at the site, and expected engineered materials [ENFE 1.15] (NRC 1998f).

- Accepted and well-documented procedures used to construct and test the numerical models for simulating coupled THC effects on seepage and flow were appropriately adopted [ENFE 1.16] (NRC 1998f).

- Abstracted models for coupled THC effects on seepage and flow were based on the same assumptions and approximations shown to be appropriate for closely analogous natural or experimental systems. Abstract model results were verified through comparison to outputs of detailed process models and empirical observations. Abstracted model results were compared with different mathematical models to judge robustness of results [ENFE 1.17] (NRC 1998f).

- Evidence is provided that all relevant FEPs have been considered. Abstracted models adequately incorporate important design features, physical phenomena, and couplings, and consistent and appropriate assumptions are used throughout [ENFE 1.18] (NRC 1998f).

- Models have reasonably accounted for known temporal and spatial variations in conditions affecting coupled THC effects on seepage and flow [ENFE 1.19] (NRC 1998f).

- Not all THC couplings may be determined to be important to performance, and the DOE may adopt assumptions to simplify PA analyses. If potentially important couplings are neglected, the DOE should provide a technical basis for doing so. The technical basis can include activities such as independent modeling, laboratory or field data, or sensitivity studies [ENFE 1.20] (NRC 1998f).

- Where simplifications were used for modeling coupled THC effects on seepage and flow in PA analyses instead of detailed process models, document and justify the bases used for modeling assumptions and approximations [ENFE 1.21] (NRC 1998f).

- Data were collected, developed, and documented under acceptable QA procedures, and models were developed and documented under acceptable QA procedures [ENFE 1.22] (NRC 1998f).

- Deficiency reports concerning data quality on issues related to coupled THC effects on seepage and flow were closed [ENFE 1.23] (NRC 1998f).

- Where data are not reasonably or practicably obtained, demonstrate that expert judgement has been used and expert elicitation procedures have been adequately

documented. If used, demonstrate that expert elicitations were conducted and documented in accordance with the requirements of the QARD (DOE 1998), [ENFE 1.24] (NRC 1998f).

8.3.1.4 Waste Package Degradation

Describe in detail the WP degradation model, including discussion of important design factors, environmental factors, and degradation processes. Discuss the implementation of the WP degradation model in the PA model and the evaluation of key aspects of WP degradation relative to performance. Describe important couplings with other models, major assumptions, sensitivities, and uncertainties. Ensure the requirements delineated in Section 8.3.1 have been addressed, as applicable.

Ensure the following programmatic criteria have been addressed (NRC 1998a). As appropriate, provide reference to supporting documentation, including other LA chapters and process model reports.

- The collection and documentation of data, as well as development and documentation of analyses, methods, models, and codes, were accomplished under approved QA and control procedures and standards [CLST 1] (NRC 1998a).

- Where data are not reasonably or practicably obtained, demonstrate that expert judgment has been used and expert elicitation procedures have been adequately documented. If used, demonstrate that expert elicitations were conducted and documented in accordance with the requirements of the QARD (DOE 1998), [CLST 2] (NRC 1998a).

- Sufficient data (field, laboratory, and natural analog) are available to adequately define relevant parameters for the models used to evaluate performance aspects of the subissues [CLST 3] (NRC 1998a).

- Sensitivity and uncertainty analyses, including consideration of alternative conceptual models, were used to determine whether additional data would be needed to better define ranges of input parameters [CLST 4] (NRC 1998a).

- Parameter values, assumed ranges, test data, probability distributions, and bounding assumptions used in the models are technically defensible and can reasonably account for known uncertainties [CLST 5] (NRC 1998a).

- Mathematical model limitations and uncertainties in modeling were defined and documented [CLST 6] (NRC 1998a).

- Primary and alternative modeling approaches consistent with available data and current scientific understanding were investigated, and their results and limitations were considered in evaluating the subissue [CLST 7] (NRC 1998a).

- Model outputs were validated through comparisons with outputs of detailed process models, empirical observations, or both [CLST 8] (NRC 1998a).

- The structure and organization of process and abstracted models were found to adequately incorporate important design features, physical phenomena, and coupled processes [CLST 9] (NRC 1998a).

Ensure the following criteria, which relate to the effects of corrosion processes on the lifetime of containers, have been addressed (NRC 1998a). As appropriate, provide reference to supporting documentation, including other LA chapters and process model reports.

- The broad range of environmental conditions within the WP emplacement drifts that may promote corrosion processes have been identified. The possibility of irregular wet and dry cycles that may enhance the rate of container degradation have been taken into account [CLST 1.2] (NRC 1998a).

- Corrosion models are adequate representations of expected container performance, and the models are unlikely to underestimate the actual performance of the containers in the repository environment [CLST 1.3] (NRC 1998a).

Ensure the following criteria, which relate to the effects of materials stability and mechanical failure on the lifetime of containers, have been addressed (NRC 1998a). As appropriate, provide reference to supporting documentation, including other LA chapters and process model reports.

- Demonstrate that the numerical models used for container material stability and mechanical failures are effective representations, taking into consideration associated uncertainties of the expected materials behavior, and are not likely to underestimate the actual rate of failure in the repository environment [CLST 2.3] (NRC 1998a).

- The compatibility of container materials and the variability of container manufacturing processes, including welding, has been considered in WP failure analyses and in the evaluation of radionuclide release [CLST 2.4] (NRC 1998a).

Ensure that the programmatic and technical criteria delineated in Section 8.3.1 have been addressed with regard to WP corrosion [TSPAI 1.1.1] (NRC 1998b). As appropriate, provide reference to supporting documentation, including other LA chapters and process model reports.

Ensure that the criteria, which relate to the effects of coupled THC processes on the WP chemical environment, have been addressed (NRC 1998f). As appropriate, provide reference to supporting documentation, including other LA chapters and process model reports.

- Available data relevant to both temporal and spatial variations in conditions affecting coupled THC effects on the WP chemical environment were considered [ENFE 2.1] (NRC 1998f).

- The evaluation of coupled THC processes properly considered site characteristics in establishing initial and boundary conditions for conceptual models and simulations of coupled processes that may affect the WP chemical environment [ENFE 2.2] (NRC 1998f).

- Sufficient data were collected on the characteristics of the natural system and engineered materials, such as the type, quantity, and reactivity of material, to establish initial and boundary conditions for conceptual models and simulations of THC coupled processes that affect the WP chemical environment [ENFE 2.3] (NRC 1998f).

- A nutrient and energy inventory calculation was used to determine the potential for microbial activity that could impact the WP chemical environment [ENFE 2.4] (NRC 1998f).

- Should microbial activity be sufficient to evaluate microbial-influenced corrosion of the WP, then the time-history of temperature, humidity, and dripping should be used to constrain its probability [ENFE 2.5] (NRC 1998f).

- Sensitivity and uncertainty analyses, including consideration of alternative conceptual models, were used to determine whether additional new data were needed to better define ranges of input parameters [ENFE 2.6] (NRC 1998f).

- Reasonable or conservative ranges of parameters or functional relations were used to determine effects of coupled THC processes on the WP chemical environment. Parameter values, assumed ranges, probability distributions, and bounding assumptions are technically defensible and reasonably account for uncertainties [ENFE 2.8] (NRC 1998f).

- Uncertainty in data due to both temporal and spatial variations in conditions affecting coupled THC effects on the WP chemical environment were considered [ENFE 2.9] (NRC 1998f).

- The evaluation of coupled THC processes properly considered the uncertainties in the characteristics of the natural system and engineered materials, such as the type, quantity, and reactivity of material, in establishing initial and boundary conditions for conceptual models and simulations of THC coupled processes that affect the WP chemical environment [ENFE 2.10] (NRC 1998f).

- The initial conditions, boundary conditions, and computational domain used in the sensitivity analyses involving coupled THC effects on the WP chemical environment were consistent with available data [ENFE 2.11] (NRC 1998f).

- Appropriate models, tests, and analyses that are sensitive to the THC couplings under consideration for both natural and engineering systems as described in the following examples were used. The effects of THC coupled processes that may occur in the natural setting or due to interactions with engineered materials or their alteration products include [ENFE 2.13] (NRC 1998f):

  - TH effects on gas and water chemistry

  - Hydrothermally driven geochemical reactions, such as zeolitization of volcanic glass, which could affect water chemistry and WP environmental conditions

- Dehydration of hydrous phases liberating moisture that may affect the WP environment

- The effects of microbial process on the WP environment

- Changes in water chemistry that may result from the release of corrosion products from the WP and interactions between cementitious materials and groundwater, which, in turn, may affect the WP chemical environment.

- Alternative modeling approaches consistent with available data and current scientific understanding were investigated, and their results and limitations appropriately considered [ENFE 2.14] (NRC 1998f).

- A reasonable description of the mathematical models included in the analysis of coupled THC effects on the WP chemical environment was provided. The description should include a discussion of alternative modeling approaches not considered in the final analysis and the limitations and uncertainties of the chosen model [ENFE 2.15] (NRC 1998f).

- The mathematical models for the WP chemical environment are consistent with conceptual models based on inferences about the near-field environment, field data and natural alteration observed at the site, and expected engineered materials [ENFE 2.16] (NRC 1998f).

- Accepted and well-documented procedures were adopted to construct and test the numerical models used to simulate the WP chemical environment [ENFE 2.17] (NRC 1998f).

- Abstracted models for coupled THC effects on the WP chemical environment were based on the same assumptions and approximations shown to be appropriate for closely analogous natural or experimental systems. Abstracted model results were verified through comparison to outputs of detailed process models and empirical observations. Abstracted model results were compared with different mathematical models to judge robustness of results [ENFE 2.18] (NRC 1998f).

- All the relevant FEPs were considered. The abstracted models adequately incorporated important design features, physical phenomena, and couplings, and used consistent and appropriate assumptions throughout [ENFE 2.19] (NRC 1998f).

- Models reasonably accounted for known temporal and spatial variations in conditions affecting coupled THC effects on the WP chemical environment [ENFE 2.20] (NRC 1998f).

- Not all THC couplings may be determined to be important to performance, and assumptions may be adopted to simplify PA analyses. If potentially important couplings are neglected, provide a technical basis for doing so. The technical basis can include

activities such as independent modeling, laboratory or field data, or sensitivity studies [ENFE 2.21] (NRC 1998f).

- Where simplifications for modeling coupled THC effects on the WP chemical environment were used for PA analyses instead of detailed process models, the bases used for modeling assumptions and approximations were documented and justified [ENFE 2.22] (NRC 1998f).

- Data and models were collected, developed, and documented under acceptable QA procedures [ENFE 2.23] (NRC 1998f).

- Deficiency reports concerning data quality on issues related to coupled THC effects on the WP chemical environment were closed [ENFE 2.24] (NRC 1998f).

- Where data are not reasonably or practicably obtained, demonstrate that expert judgment has been used and expert elicitation procedures have been adequately documented. If used, demonstrate that expert elicitations were conducted and documented in accordance with the requirements of the QARD (DOE 1998) [ENFE 2.25] (NRC 1998f).

8.3.1.5 Waste Form Degradation

Describe in detail the conceptual models for waste form degradation. Include discussions of the rate of degradation of cladding and the waste matrix, the rate of mobilization of radioisotopes, and the migration of radioisotopes through the remaining portions of the WP. Describe the important couplings with other models, major assumptions, sensitivities, and uncertainties. Ensure the requirements delineated in Section 8.3.1 have been addressed, as applicable.

Ensure that the programmatic and technical criteria delineated in Section 8.3.1 have been addressed with regard to radionuclide release rates and solubility limits [TSPAI 1.1.4] (NRC 1998b). As appropriate, provide reference to supporting documentation, including other LA chapters and process model reports.

Ensure that the following criteria for the rate of degradation of spent nuclear fuel (SNF) have been addressed (NRC 1998a). As appropriate, provide reference to supporting documentation, including other LA chapters and process model reports.

- Identify and consider likely processes for degradation of SNF and the release of radionuclides from the EBS. Discuss dissolution of the irradiated UO_2 matrix and the formation of secondary minerals and colloids, prompt release of radionuclides, degradation in the dry air environment, degradation and failure of fuel cladding, preferential dissolution of intermetallics in DOE SNF, and the release of radionuclides from the WP emplacement drifts [CLST 3.2] (NRC 1998a).

- Demonstrate that the numerical models used to assess SNF degradation and radionuclide release from the EBS are adequate representations, including consideration of uncertainties, of the expected SNF performance and that the models are unlikely to

overestimate the actual performance in the repository environment [CLST 3.3] (NRC 1998a).

Ensure that the following criteria for the rate of degradation of high-level waste glass have been addressed (NRC 1998a). As appropriate, provide reference to supporting documentation, including other LA chapters and process model reports.

- Demonstrate that the likely processes for degradation of high-level radioactive waste (HLW) glass and the release of radionuclides from the EBS have been identified and considered. This may include dissolution of the primary phase, formation of secondary minerals and colloids, microbial action, and radionuclide releases and transport from the WP emplacement drift [CLST 3.2] (NRC 1998a).

- Demonstrate that the numerical models used for determining the rate of dissolution of HLW glass and the rate of radionuclide release from the EBS are adequate representations, taking into consideration the associated uncertainties, of the expected performance of the HLW glass, and that the results of these models are unlikely to underestimate the actual rate of degradation of the HLW glass and the subsequent rate of release in the repository environment [CLST 3.3] (NRC 1998a).

- Assess the compatibility of HLW glass with internal components of the WP in the evaluation of radionuclide release, taking into consideration co-disposal with SNF owned by the DOE in the same WP. Specifically, HLW glass should not compromise the performance of the WP [CLST 3.4] (NRC 1998a).

Ensure that the criteria that deal with the effects of coupled THC processes on the chemical environment for radionuclide release have been addressed (NRC 1998f). As appropriate, provide reference to supporting documentation, including other LA chapters and process model reports.

- Show that available data relevant to both temporal and spatial variations in conditions affecting coupled THC effects on the chemical environment for radionuclide release were considered [ENFE 3.1] (NRC 1998f).

- Demonstrate that, during evaluation of the coupled THC processes, site characteristics were properly considered in establishing initial and boundary conditions for conceptual models and simulations of the coupled processes that may affect the chemical environment for radionuclide release [ENFE 3.2] (NRC 1998f).

- Demonstrate that sufficient data on the characteristics of the natural system and engineered materials (e.g., type, quantity, and reactivity of material) were collected to establish initial and boundary conditions for conceptual models and simulations of the THC coupled processes that affect the chemical environment for radionuclide release [ENFE 3.3] (NRC 1998f).

- Provide a nutrient and energy inventory calculation to determine the potential for microbial activity that could adversely affect radionuclide release [ENFE 3.4] (NRC 1998f).

- Should microbial activity be sufficient to potentially affect the chemical environment for radionuclide release, then the time-history of temperature, humidity, and dripping should be used to constrain the probability of microbial effects (e.g., production of organic by-products) that act as complexing ligands for actinides and microbially enhanced dissolution of the HLW glass form [ENFE 3.5] (NRC 1998f).

- Ensure that sensitivity and uncertainty analyses, including consideration of alternative conceptual models, were conducted to determine whether additional new data are needed to better define the range of values used for the input parameters [ENFE 3.6] (NRC 1998f).

- Reasonable or conservative ranges of parameters or functional relations were used to determine effects of coupled THC processes on the WP chemical environment. Parameter values, assumed ranges, probability distributions, and bounding assumptions are technically defensible and reasonably account for uncertainties [ENFE 3.8] (NRC 1998f).

- Ensure that uncertainty in the data (due to temporal and spatial variations in conditions affecting coupled THC effects on the chemical environment for radionuclide release) was considered [ENFE 3.9] (NRC 1998f).

- Demonstrate that the DOE evaluation of the coupled THC processes properly considered the uncertainties in the characteristics of the natural system and engineered materials (e.g., type, quantity, and reactivity of material) in establishing initial and boundary conditions for conceptual models and simulations of THC coupled processes that affect the chemical environment for radionuclide release [ENFE 3.10] (NRC 1998f).

- The initial conditions, boundary conditions, and computational domain that were used in the sensitivity analyses involving coupled THC effects on the chemical environment for radionuclide release were consistent with available data [ENFE 3.11] (NRC 1998f).

- Appropriate models, tests, and analyses (i.e., those that are sensitive to the THC couplings under consideration for the natural and engineering systems) were used [ENFE 3.13] (NRC 1998f).

- Alternative modeling approaches consistent with available data and current scientific understanding were investigated, and their results and limitations appropriately considered [ENFE 3.14] (NRC 1998f).

- Provide a reasonable description of the mathematical models that were used in the analysis of coupled THC effects on the chemical environment for radionuclide release. Include a discussion of alternative modeling approaches (those not considered in the final analysis) and the limitations and uncertainties of the chosen model [ENFE 3.15] (NRC 1998f).

- Ensure that mathematical models used for modeling coupled THC effects on the chemical environment for radionuclide release are consistent with conceptual models

based on inferences about the near-field environment, field data and natural alteration observed at the site, and expected engineered materials [ENFE 3.16] (NRC 1998f).

- Ensure that the DOE appropriately adopted accepted and well-documented procedures to construct and test the numerical models used to simulate coupled THC effects on the chemical environment for radionuclide release [ENFE 3.17] (NRC 1998f).

- Ensure that abstracted models of coupled THC effects on the chemical environment for radionuclide release were based on the same assumptions and approximations shown to be appropriate for closely analogous natural or experimental systems. Ensure that abstracted model results were verified through comparison to outputs of detailed process models and empirical observations. Ensure that abstracted model results are compared with different mathematical models to judge robustness of results [ENFE 3.18] (NRC 1998f).

- Ensure that all the relevant FEPs have been considered. Ensure that the abstracted models adequately incorporate important design features, physical phenomena, and couplings, and that consistent and appropriate assumptions were used throughout [ENFE 3.19] (NRC 1998f).

- Models reasonably accounted for known temporal and spatial variations in conditions affecting coupled THC effects on the WP chemical environment [ENFE 3.20] (NRC 1998f).

- Not all THC couplings may be determined to be important to performance, and assumptions may be adopted to simplify PA analyses. If potentially important couplings are neglected, provide a technical basis for doing so. The technical basis can include activities such as independent modeling, laboratory or field data, or sensitivity studies [ENFE 3.21] (NRC 1998f).

- Where simplifications were used for modeling coupled THC effects on the chemical environment for radionuclide release in PA analyses instead of detailed process models, document and justify the bases used for modeling assumptions and approximations [ENFE 3.22] (NRC 1998f).

- Data and models were collected, developed, and documented under acceptable QA procedures [ENFE 3.23] (NRC 1998f).

- Ensure that deficiency reports concerning data quality on issues related to coupled THC effects on the chemical environment for radionuclide release were closed [ENFE 3.24] (NRC 1998f).

- Where data are not reasonably or practicably obtained, demonstrate that expert judgment has been used and expert elicitation procedures have been adequately documented. If used, demonstrate that expert elicitations were conducted and documented in accordance with the requirements of the QARD (DOE 1998) [ENFE 3.25] (NRC 1998f).

8.3.1.6 Engineered Barrier System Transport

Describe in detail the conceptual models for the transport of materials through the EBS model. Include a discussion of the implementation of the EBS model in the PA model, model inputs and outputs, components of the model, and evaluation of key aspects of the subsystem relative to performance. Describe important couplings with other models, major assumptions, sensitivities, and uncertainties. Ensure that the requirements delineated in Section 8.3.1 have been addressed, as applicable.

Ensure that the criteria, which relate to the effects of coupled THC processes on radionuclide transport through engineered and natural barriers, have been addressed (NRC 1998f). As appropriate, provide reference to supporting documentation including other LA chapters and process model reports.

- Available data relevant to both temporal and spatial variations in conditions affecting coupled THC effects on the WP chemical environment were considered [ENFE 4.1] (NRC 1998f).

- The evaluation of coupled THC processes properly considered site characteristics in establishing initial and boundary conditions for conceptual models and simulations of coupled processes that may affect the WP chemical environment [ENFE 4.2] (NRC 1998f).

- Sufficient data were collected on the characteristics of the natural system and engineered materials, such as the type, quantity, and reactivity of material, to establish initial and boundary conditions for conceptual models and simulations of THC coupled processes that affect the WP chemical environment [ENFE 4.3] (NRC 1998f).

- A nutrient and energy inventory calculation was used to determine the potential for microbial activity that could impact the WP chemical environment [ENFE 4.4] (NRC 1998f).

- Should microbial activity be sufficient to evaluate microbially influenced corrosion of the WP, then the time-history of temperature, humidity, and dripping should be used to constrain its probability [ENFE 4.5] (NRC 1998f).

- Sensitivity and uncertainty analyses, including consideration of alternative conceptual models, were used to determine whether additional new data were needed to better define ranges of input parameters [ENFE 4.6] (NRC 1998f).

- Reasonable or conservative ranges of parameters or functional relations were used to determine effects of coupled THC processes on the WP chemical environment. Parameter values, assumed ranges, probability distributions, and bounding assumptions are technically defensible and reasonably account for uncertainties [ENFE 4.8] (NRC 1998f).

- Uncertainty in data due to both temporal and spatial variations in conditions affecting coupled THC effects on the WP chemical environment were considered [ENFE 4.9] (NRC 1998f).

- The evaluation of coupled THC processes properly considered the uncertainties in the characteristics of the natural system and engineered materials, such as the type, quantity, and reactivity of material, in establishing initial and boundary conditions for conceptual models and simulations of THC coupled processes that affect the WP chemical environment [ENFE 4.10] (NRC 1998f).

- The initial conditions, boundary conditions, and computational domain used in the sensitivity analyses involving coupled THC effects on the WP chemical environment were consistent with available data [ENFE 4.11] (NRC 1998f).

- Appropriate models, tests, and analyses that are sensitive to the THC couplings under consideration for both natural and engineering systems as described in the following examples were used. The effects of THC coupled processes that may occur in the natural setting or due to interactions with engineered materials or their alteration products include [ENFE 4.13] (NRC 1998f):

  - TH effects on gas and water chemistry

  - Hydrothermally driven geochemical reactions, such as zeolitization of volcanic glass, which could affect water chemistry and WP environmental conditions

  - Dehydration of hydrous phases liberating moisture that may affect the WP environment

  - Effects of microbial process on the WP environment

  - Changes in water chemistry that may result from the release of corrosion products from the WP and interactions between cementitious materials and groundwater, which, in turn, may affect the WP chemical environment.

- Alternative modeling approaches consistent with available data and current scientific understanding were investigated, and their results and limitations appropriately considered [ENFE 4.14] (NRC 1998f).

- A reasonable description of the mathematical models included in the analysis of coupled THC effects on the WP chemical environment was provided. The description should include a discussion of alternative modeling approaches not considered in the final analysis and the limitations and uncertainties of the chosen model [ENFE 4.15] (NRC 1998f).

- The mathematical models for the WP chemical environment are consistent with conceptual models based on inferences about the near-field environment, field data and natural alteration observed at the site, and expected engineered materials [ENFE 4.16] (NRC 1998f).

- Accepted and well-documented procedures were adopted to construct and test the numerical models used to simulate the WP chemical environment [ENFE 4.17] (NRC 1998f).

- Abstracted models for coupled THC effects on the WP chemical environment were based on the same assumptions and approximations shown to be appropriate for closely analogous natural or experimental systems. Abstracted model results were verified through comparison to outputs of detailed process models and empirical observations. Abstracted model results were compared with different mathematical models to judge robustness of results [ENFE 4.18] (NRC 1998f).

- All the relevant FEPs were considered. The abstracted models adequately incorporated important design features, physical phenomena, and couplings, and used consistent and appropriate assumptions throughout [ENFE 4.19] (NRC 1998f).

- Models reasonably accounted for known temporal and spatial variations in conditions affecting coupled THC effects on the WP chemical environment [ENFE 4.20] (NRC 1998f).

- Not all THC couplings may be determined to be important to performance, and assumptions may be adopted to simplify PA analyses. If potentially important couplings are neglected, provide a technical basis for doing so. The technical basis can include activities such as independent modeling, laboratory or field data, or sensitivity studies [ENFE 4.21] (NRC 1998f).

- Where simplifications for modeling coupled THC effects on the WP chemical environment were used for PA analyses instead of detailed process models, the bases used for modeling assumptions and approximations were documented and justified [ENFE 4.22] (NRC 1998f).

- Data and models were collected, developed, and documented under acceptable QA procedures [ENFE 4.23] (NRC 1998f).

- Deficiency reports concerning data quality on issues related to coupled THC effects on the WP chemical environment were closed [ENFE 4.24] (NRC 1998f).

- Where data are not reasonably or practicably obtained, demonstrate that expert judgment has been used and expert elicitation procedures have been adequately documented. If used, demonstrate that expert elicitations were conducted and documented in accordance with the requirements of the QARD (DOE 1998), [ENFE 4.25] (NRC 1998f).

8.3.1.7 Saturated Zone Flow and Transport

Describe the conceptual models of the flow and transport processes in the SZ relevant to radionuclide migration from the repository and show how the model has been applied. Include a description of the construction of the conceptual models, implementation of the models, and results and interpretation of the analyses. Describe important couplings with other models,

major assumptions, sensitivities, and uncertainties. Ensure that the requirements delineated in Section 8.3.1 have been addressed, as applicable.

Ensure that the programmatic and technical criteria delineated in Section 8.3.1 have been addressed with regard to flow rates in water production zones [TSPAI 2.2.1] (NRC 1998b). As appropriate, provide reference to supporting documentation, including other LA chapters and process model reports.

Ensure that the programmatic and technical criteria delineated in Section 8.3.1 have been addressed with regard to retardation flow rates in water production zones [TSPAI 2.2.2] (NRC 1998b). As appropriate, provide reference to supporting documentation, including other LA chapters and process model reports.

8.3.1.8 Biosphere

Describe incorporation of the biosphere models and calculations into the PA. Include a description of the model of the critical group and the biosphere pathways that might direct radionuclides to the critical group. The biosphere dose conversion factors should be discussed. Also, describe the calculations used to predict the annual radiation dose to the reference person. Include a description of the construction of the conceptual models, implementation of the models, and results and interpretation of the analyses. Describe important couplings with other models, major assumptions, sensitivities, and uncertainties. Ensure that the requirements delineated in Section 8.3.1 have been addressed, as applicable.

Explain how FEPs describing the reference biosphere are consistent with present knowledge of the conditions in the Yucca Mountain region (proposed 10 CFR 63.115(a)(1) [64 FR 8640]). Explain that reliance on present knowledge and conditions is considered reasonable for development of exposure scenarios because such exposure scenarios can be based on empirical knowledge rather than unconstrained speculation. Refer to Section 3.1 for additional details regarding the geography of the region around Yucca Mountain.

Explain how the biosphere pathways are consistent with arid or semi-arid conditions as described in the technical bases for the biosphere model (proposed 10 CFR 63.115(a)(2) [64 FR 8640]). Explain that, based on current interpretations of fossil records, paleoclimate studies, SZ flow model, and climate change models, it is reasonable to limit the assumed climate change to these possibilities.

Ensure that the programmatic and technical criteria delineated in Section 8.3.1 have been addressed with regard to dilution of radionuclides in groundwater due to well pumping [TSPAI 3.1.1] (NRC 1998b). As appropriate, provide reference to supporting documentation, including other LA chapters and process model reports.

Ensure that the programmatic and technical criteria delineated in Section 8.3.1 have been addressed with regard to dilution of radionuclides in soil due to surface properties [TSPAI 3.1.2] (NRC 1998b). As appropriate, provide reference to supporting documentation, including other LA chapters and process model reports.

8.3.2 Integration of Subsystem Models

Discuss the approach for combining the component models into an overall model and computer code. Describe the information flow between models and the computer code architecture that facilitates the information flow. This description should describe how to recouple the component models into one integral whole, reassemble the analyzed pieces, and pass information between them to develop reasonable assessments of overall system performance.

8.3.3 Models and Data for Disruptive Events

Describe in detail the conceptual models for the disruptive event scenarios that have been identified for further analysis (as discussed in Section 8.2.2). Include a description of the construction of the conceptual models, implementation of the models, and results and interpretation of the analyses. Ensure that the technical and programmatic requirements delineated in Section 8.3.1 have been addressed, as applicable.

Ensure that the programmatic and technical criteria delineated in Section 8.3.1 have been addressed with regard to mechanical disruption of WPs [TSPAI 1.1.2] (NRC 1998b). As appropriate, provide reference to supporting documentation, including other LA chapters and process model reports.

Ensure that the programmatic and technical criteria delineated in Section 8.3.1 have been addressed with regard to volcanic disruption of WPs [TSPAI 2.3.1] (NRC 1998b). As appropriate, provide reference to supporting documentation, including other LA chapters and process model reports.

Ensure that the following criteria, which deal with the effects of seismically induced rockfall on WP performance, have been addressed (NRC 1998c). As appropriate, provide reference to supporting documentation, including other LA chapters and process model reports.

- Show that the seismic hazard inputs used to estimate rockfall potential are consistent with the inputs used in the design and performance assessments, as established in the topical report addressing design and PA inputs [RDTME 3.2.3] (NRC 1998c).

- Give evidence that the size distribution of rocks that may potentially fall on the WPs is estimated from site-specific data (e.g., distribution of joint patterns, spacing, and orientation in three dimensions) with adequate consideration of uncertainties [RDTME 3.2.4] (NRC 1998c).

- Show that the analytical model used in the estimation of impact loads due to rockfall is based on reasonable assumptions and site data, is consistent with the emplacement drift and WP designs, and is defensible with respect to providing realistic or bounding estimates of impact loads and stresses [RDTME 3.2.5] (NRC 1998c).

- Discuss how the PA implements the results of rockfall analyses addressing the following: (1) the possibility of multiple blocks simultaneously falling onto a WP, (2) the extent of potential rockfall area around an individual emplacement drift as a

function of ground motion, and (3) the extent of potential rockfall area over the entire repository as functions of ground motion [RDTME 3.2.7] (NRC 1998c).

Ensure that the following criteria, which deal with the effects of the WP and EBS designs on the potential for nuclear criticality, have been addressed (NRC 1998a). As appropriate, provide reference to supporting documentation including other LA chapters and process model reports.

- The DOE has developed conceptual models for evaluating criticality safety and for determining quantitative values for the effective neutron multiplication factor (k_{eff}), including appropriate biases and uncertainties in the models [CLST 5.1] (NRC 1998a).

- The DOE has evaluated the potential for nuclear criticality occurring inside breached WPs by appropriate consideration of different types of SNF and vitrified waste while taking into account the design of the WP and other EBS components and establishing adequate initial and boundary conditions for conceptual models and simulations [CLST 5.2] (NRC 1998a).

Ensure that the criteria, which relate to the effects of coupled THC processes affecting criticality in the near-field, have been addressed (NRC 1998f). As appropriate, provide reference to supporting documentation, including other LA chapters and process model reports.

- Sensitivity and uncertainty analyses, including consideration of alternative conceptual models, were used to determine whether criticality will impact repository performance, and whether additional new data are needed to better define ranges of input parameters [ENFE 5.1] (NRC 1998f).

- Available data relevant to both temporal and spatial variations in conditions affecting coupled THC effects on the potential for nuclear criticality in the near-field environment were considered [ENFE 5.2] (NRC 1998f).

- The evaluation of coupled THC processes properly considered site characteristics in establishing initial and boundary conditions for conceptual models and simulations of coupled processes that may affect nuclear criticality in the near-field environment [ENFE 5.3] (NRC 1998f).

- Sufficient data were collected on the characteristics of the natural system and engineered materials, such as the type, quantity, and reactivity of material, to establish initial and boundary conditions for conceptual models and simulations of THC coupled processes that may affect nuclear criticality in the near-field environment [ENFE 5.4] (NRC 1998f).

- Reasonable or conservative ranges of parameters or functional relations were used to determine effects of coupled THC processes on potential nuclear criticality in the near-field environment. Parameter values, assumed ranges, probability distributions, and bounding assumptions are technically defensible and reasonably account for uncertainties [ENFE 5.5] (NRC 1998f).

- Uncertainty in data due to both temporal and spatial variations in conditions affecting coupled THC effects on potential nuclear criticality were considered [ENFE 5.6] (NRC 1998f).

- The evaluation of coupled THC processes properly considered the uncertainties in the characteristics of the natural system and engineered materials, such as the type, quantity, and reactivity of material, in establishing initial and boundary conditions for conceptual models and simulations of THC coupled processes that affect potential nuclear criticality [ENFE 5.7] (NRC 1998f).

- The initial conditions, boundary conditions, and computational domain used in the sensitivity analyses involving coupled THC effects on potential nuclear criticality in the near-field environment were consistent with available data [ENFE 5.8] (NRC 1998f).

- Alternative modeling approaches consistent with available data and current scientific understanding were investigated, and their results and limitations appropriately considered [ENFE 5.9] (NRC 1998f).

- A reasonable description of the mathematical models included in the analysis of coupled THC effects on potential nuclear criticality was provided. The description should include a discussion of alternative modeling approaches not considered in the final analysis and the limitations and uncertainties of the chosen model [ENFE 5.10] (NRC 1998f).

- The mathematical models for potential nuclear criticality are consistent with conceptual models based on inferences about the near-field environment, field data and natural alteration observed at the site, and expected engineered materials [ENFE 5.11] (NRC 1998f).

- Accepted and well-documented procedures were adopted to construct and test the numerical models used to simulate coupled THC effects on potential nuclear criticality [ENFE 5.12] (NRC 1998f).

- Abstracted models for coupled THC effects on potential nuclear criticality were based on the same assumptions and approximations shown to be appropriate for closely analogous natural or experimental systems. Abstracted model results were verified through comparison to outputs of detailed process models and empirical observations. Abstracted model results were compared with different mathematical models to judge robustness of results [ENFE 5.13] (NRC 1998f).

- All the relevant FEPs were considered. The abstracted models adequately incorporated important design features, including criticality safety features, physical phenomena, and couplings, and used consistent and appropriate assumptions throughout [ENFE 5.14] (NRC 1998f).

- Important mass transfer and mass transport processes and mechanisms considered for formation of both a critical mass and configuration are plausible for the Yucca Mountain near-field environment [ENFE 5.15] (NRC 1998f).

- Models reasonably accounted for known temporal and spatial variations in conditions affecting coupled THC effects on potential nuclear criticality [ENFE 5.16] (NRC 1998f).

- Criticality in the near field, and not all THC couplings, may be determined to be important to performance, and the DOE may adopt assumptions to simplify PA analyses. If potentially important couplings and criticality in the near field are neglected, the DOE should provide a technical basis for doing so. The technical basis could include activities, such as independent modeling, laboratory or field data, or sensitivity studies [ENFE 5.17] (NRC 1998f).

- Where simplifications for modeling coupled THC effects on potential nuclear criticality were used for PA analyses instead of detailed process models, the bases used for modeling assumptions and approximations were documented and justified [ENFE 5.18] (NRC 1998f).

- Data and models were collected, developed, and documented under acceptable QA procedures [ENFE 5.19] (NRC 1998f).

- Deficiency reports concerning data quality on issues related to coupled THC effects on the potential nuclear criticality were closed [ENFE 5.20] (NRC 1998f).

- Where data are not reasonably or practicably obtained, demonstrate that expert judgment has been used and expert elicitation procedures have been adequately documented. If used, demonstrate that expert elicitations were conducted and documented in accordance with the requirements of the QARD (DOE 1998) [ENFE 5.21] (NRC 1998f).

8.4 TOTAL SYSTEM PERFORMANCE ASSESSMENT

Provide a brief discussion stating that the purpose of this section is to present the results of the evaluation of the nominal and disruptive scenarios. Explain that the overall results of the TSPA without human intrusion are also presented.

8.4.1 Evaluation of the Nominal Scenario

State the purpose of this section is to describe and summarize the ability of the geologic repository to limit radiologic exposures to those specified in proposed 10 CFR 63.113(b) (64 FR 8640), which is demonstrated through a PA that meets the requirements specified in Interim Guidance Section 114 (Dyer and Horton 1999), uses the reference biosphere and critical group specified in Interim Guidance Section 115 (Dyer and Horton 1999), and excludes the effects of human intrusion (proposed 10 CFR 63.113(c) [64 FR 8640]).

Describe the objectives of the PA and discuss the approach used to analyze the potential Yucca Mountain repository system. Describe in detail how the potential repository system is represented in the TSPA. Describe the components of the overall waste disposal system. Briefly

describe the key aspects of the individual components identified for the TSPA. Describe how the key aspects of the individual components were identified.

Describe how the PA is based on the assumption that evolution of the Yucca Mountain geologic setting over the postclosure performance period will be consistent with present knowledge of natural processes as detailed in Chapter 3. Provide reference, as appropriate, to justify the assertions being made (Interim Guidance Section 114(l) [Dyer and Horton 1999]).

Describe the results of the assessment of the degree to which FEPs are expected to materially affect compliance with proposed 10 CFR 63.113(b) (64 FR 8640), whether potentially beneficial or adverse to performance of the geologic repository. State that investigations extend from the surface to a depth sufficient to determine principal pathways for radionuclide migration from the underground facility (proposed 10 CFR 63.21(c)(5) [64 FR 8640]). This ensures consideration for inclusion of FEPs with both beneficial and adverse effects on performance.

Summarize the key parameters of the components, models, and abstractions described in Section 8.3 that are most responsible for variations in performance. Provide a concise summary of the parameter values, models, and abstractions used in the TSPA.

Describe how complete assurance that the performance objectives will be met is not expected. Explain that a reasonable assurance, on the basis of the record before the NRC, that the performance objectives will be met is the standard. Explain that the TSPA described in this chapter supports the NRC finding of reasonable assurance (proposed 10 CFR 63.101(a)(2) [64 FR 8640]).

Describe how demonstrating compliance will involve the use of complex models that are supported by limited data from field and laboratory tests, site-specific monitoring and natural analog studies, and that output from the models may be supplemented with expert judgment. Explain that the analyses and models that will be used to assess performance of the geologic repository will be supported by using an appropriate combination of such methods as field tests, in situ tests, laboratory tests that represent field conditions, monitoring data, and natural analog studies (proposed 10 CFR 63.21(c)(9) [64 FR 8640]). Describe how uncertainties and gaps in knowledge inherent in PA are compensated with bounding assumptions and conservative approaches. Briefly describe how the multiple barrier approaches also add confidence that may be factored into the NRC reasonable assurance finding.

Describe the critical group in the accessible environment and explain how this group is used in PA calculations. Explain that the critical group is assumed to reside in a hypothetical farming community approximately 20 km south from the underground facility, in the general location of the junction of U.S. Route 95 and Nevada Route 373 (Interim Guidance Section 115(b)(1) [Dyer and Horton 1999]).

Explain that the behavior and characteristics of the assumed farming community are consistent with current conditions in the Yucca Mountain region (proposed 10 CFR 63.21(c)(1)(iv) and proposed 10 CFR 63.115(b)(2) [64 FR 8640]).

Explain that the average member of the critical group is assumed to be an adult and that metabolic and physiological considerations are consistent with present knowledge of adults (proposed 10 CFR 63.115(b)(5) [64 FR 8640]). Explain that the critical group resides within a hypothetical farming community consisting of approximately 100 individuals and exhibits behaviors or characteristics that will result in the highest expected annual doses (proposed 10 CFR 63.115(b)(3) [64 FR 8640]).

Explain that details regarding the characteristics, behaviors, and attributes of the critical group (including the mean value of the critical group's variability range, metabolic and physiological characteristics, and the methods used to establish the characteristics) will be contained in supporting documentation. Document and justify the mean value utilized, and assure that the mean value is not unduly biased because of the extreme habits of a few individuals (proposed 10 CFR 63.115(b)(4) [64 FR 8640]).

Ensure that the following criteria, which deal with the effects of alternate engineered barrier subsystem design features, are addressed [CLST 6] (NRC 1998a).

- Identify and consider the effects of backfill, and the timing of its emplacement, on the thermal loading of the repository, WP lifetime (including container corrosion and mechanical failure), and the release of radionuclides from the EBS.

- Identify and consider the effects of drip shields (with backfill) on WP lifetime, including extension of the humid-air corrosion regime, environmental effects, breakdown of drip shields and resulting mechanical impacts on WP, the potential for crevice corrosion at the junction between the WP and the drip shield, and the potential for condensate formation and dripping on the underside of the shield.

8.4.2 Evaluation of Disruptive Scenarios

State the purpose of this section, which is to provide analyses evaluating the degree to which disruptive events are expected to materially affect compliance with proposed 10 CFR 63.113(b) (64 FR 8640), taking into consideration their probability of occurrence as specified in proposed 10 CFR 63.114(d) (64 FR 8640). Describe the scenarios included and excluded from the analyses, including justifications for each category. Analyses will either model the complete processes (source term and transport through principal pathways) or limited aspects of the processes (the change in radionuclide source term), whichever is required to determine the effects on performance. Demonstrate that specific FEPs of the geologic setting outside of the site were investigated if they affected performance of the geologic repository and present the results.

Ensure that the programmatic and technical criteria delineated in Section 8.3.1 have been addressed with regard to the airborne transport of radionuclides [TSPAI 2.3.2] (NRC 1998b). As appropriate, provide reference to supporting documentation, including other LA chapters and process model reports.

Ensure that the following criteria, which relate to the consequence of igneous activities [IA 2], have been addressed (NRC 1998g).

- The models are consistent with the geologic record of basaltic igneous activity within the Yucca Mountain region [IA 2.1] (NRC 1998g).

- The models are verified against igneous processes observed at active or recently active analog igneous systems and reflect the fundamental details of ash plume dynamics [IA 2.2] (NRC 1998g).

- The models adequately account for changes in magma ascent characteristics and magma/rock interactions brought about by repository construction [IA 2.3] (NRC 1998g).

- The models account for the interactions of basaltic magma with engineered barriers and waste forms [IA 2.4] (NRC 1998g).

- The parameters are constrained by data from Yucca Mountain region igneous features and from appropriate analog systems such that the effects of igneous activity on waste containment and isolation are not underestimated [IA 2.5] (NRC 1998g).

- Where data are not reasonably or practicably obtained, demonstrate that expert judgment has been used and expert elicitation procedures have been adequately documented. If used, demonstrate that expert elicitations were conducted and documented in accordance with the requirements of the QARD (DOE 1998), [IA 2.6] (NRC 1998g).

- The collection, documentation, and development of data and models have been performed in accordance with the QARD (DOE 1998), or if data was not collected under an established QA program, it has been qualified under appropriate QA procedures.

8.4.3 Overall Results for Total System Performance without Human Intrusion

State that the purpose of this section is to summarize the TSPA analyses and results for the nominal scenario evaluations, and present a comparison against the regulatory requirements of proposed 10 CFR 63.113(b) (64 FR 8640). Describe the results of combining all of the component models (previously described) into one total system performance model. Specifically, describe the prediction of the overall model with respect to total system performance (i.e., dose rate at the accessible environment) and to subsystem performance of the various components (e.g., performance of the SZ by itself). Quantify the range of possible outcomes of overall system behavior for the total system performance. Describe in detail how the behavior of the various components influences variation in the dose and variation in the other components.

Summarize the PA of the proposed geologic repository in which disruptive events occur during the period after permanent closure. Include discussion of basaltic igneous activity, seismic activity, and nuclear criticality.

As a part of this evaluation, provide an assessment of the anticipated response of the geomechanical, hydrogeologic, and geochemical systems to the range of thermal loadings under consideration (given the pattern of fractures and other discontinuities, and the heat transfer

properties of the rock mass and groundwater) (proposed 10 CFR 63.21(c)(6) [64 FR 8640]). State that detailed descriptions of the natural characteristics of the site and the surrounding region are provided in Chapter 3.

Present results of the sensitivity analysis for the nominal scenario. Include a discussion of sensitivity analysis approaches, uncertainty analysis approaches, and the role of bounding and conservative PAs. Summarize the assessment of the sensitivity of the results to changes in the various parameters used to construct the various models. Summarize the sensitivity studies, including sensitivity of results to near-field environment uncertainties, WP and EBS performance uncertainties, gaseous flow and transport uncertainties, UZ flow and transport uncertainties, SZ flow and transport uncertainties, biosphere transport uncertainties, alternative repository thermal loads, and climate changes.

8.5 HUMAN INTRUSION ANALYSIS

This section of Chapter 8 provides a description of the stylized analysis of human intrusion. The author will produce a section that describes the technical bases for human intrusion, describes the conceptual models and data, states assumptions, and presents the results of the analysis.

8.5.1 Stylized Human Intrusion

Proposed 10 CFR 63.21(c)(8) (64 FR 8640) requires an assessment of the ability of the proposed geologic repository to limit radiological exposures in the event of limited human intrusion into the EBS. Describe how Interim Guidance Section 113(d) (Dyer and Horton 1999) requires that the ability of the geologic repository to continue to isolate waste from the accessible environment over the long term in the event of limited human intrusion into the EBS shall be analyzed, and how the results and bases of the analyses shall be presented. Describe the purpose of the analyses, which is to assess the resilience of the repository system in terms of its ability, after intrusion, to recover and continue to isolate waste from the accessible environment over the long term. This analysis will be based on a separate PA that meets the requirements specified in Interim Guidance Section 114 (Dyer and Horton 1999) and uses the reference biosphere and critical group specified in Interim Guidance Section 115 (Dyer and Horton 1999).

Explain that while no quantitative regulatory limit applies to the results, they will provide a qualitative indicator of the ability of the proposed geologic repository to continue to perform acceptably following human intrusion. Describe the stylized analysis of human intrusion scenario, including the technical bases for human intrusion analyses. Provide background information pertaining to the stylized human intrusion analysis, including a brief description of the regulatory basis and any assumptions made in the analyses.

8.5.2 Models and Data for the Human Intrusion Analysis

Describe the stylized analysis of human intrusion, including the technical bases for human intrusion analyses. Include a description of the construction of the conceptual models, implementation of the models, and the results and interpretation of the analyses. State the assumptions for the stylized analysis of human intrusion including the assumption that human intrusion occurs 100 years after permanent closure and takes the form of a drilling event that

results in a single, nearly vertical borehole that penetrates a WP, extends to the SZ, and is not adequately sealed (Interim Guidance Section 113(d) [Dyer and Horton 1999]).

Ensure all applicable requirements delineated in Section 8.3.1 for models used in this analysis and previously described have been addressed in that section. For models described in this section and not in Section 8.3.1, assure the applicable requirements of Section 8.3.1 are addressed here.

8.5.3 Evaluation of the Human Intrusion Analysis

Describe the evaluation of the ability of the potential geologic repository to continue to isolate waste from the accessible environment over the long term in the event of limited human intrusion into the EBS. Explain that no quantitative limit applies to the results. State and justify that the results will provide a qualitative indicator of the ability of the potential geologic repository to continue to perform acceptably following human intrusion. State that this evaluation is a separate analysis that uses the reference design biosphere and the critical group specified in proposed 10 CFR 63.115 (64 FR 8640).

Summarize the bases and results of the human intrusion evaluation. Explain that the post-intrusion performance of the potential repository is satisfactory if the dose rate returns, over a reasonable period of time, to a value close to the dose rate without human intrusion (Interim Guidance Section 113(d) [Dyer and Horton 1999]).

Summarize the sensitivity studies for human intrusion. Describe the sensitivity of these analyses to the key assumptions.

8.6 MULTIPLE-BARRIER ANALYSES

In this section, the concept of the multiple barrier analysis is described, features of the engineered barriers and geologic setting important to waste isolation are identified, and the effectiveness of the barriers to isolate waste and protect against the release of radioactive material to the environment are described in detail.

8.6.1 Approach to Multiple Barrier Analysis

Describe the concept of the multiple barrier analysis approach and its effectiveness in protecting against the release of radioactive material to the environment. Explain that it is intended that natural barriers and the EBS work in combination to enhance the resiliency of the geologic repository and increase confidence that the postclosure performance objective in proposed 10 CFR 63.113(b) (64 FR 8640) will be achieved.

8.6.2 Identification and Description of Barriers

Identify the design features of the EBS and the natural features of the geologic setting that are considered barriers important to waste isolation. Refer to Chapter 3 for a detailed description of the general capabilities of the natural features and processes that are expected to act as barriers. Refer to Chapter 5 for detailed descriptions of the WP, and refer to Chapter 6 for detailed descriptions of the EBS (proposed 10 CFR 63.114(h) [64 FR 8640]).

8.6.3 Results of Multiple Barrier Analysis

Describe in detail the capability of the barriers, identified as important to waste isolation, to isolate waste and protect against the release of radioactive material to the environment. Describe the uncertainties in characterizing and modeling the barriers. Also, provide the technical basis for the capability of barriers to isolate waste. Refer to Chapter 3 for a detailed description of the general capabilities of the natural features and processes that are expected to act as barriers. Refer to Chapter 5 for detailed descriptions of the WP, and refer to Chapter 6 for detailed descriptions of the EBS (proposed 10 CFR 63.114(i) and proposed 10 CFR 63.114(j) [64 FR 8640]).

8.7 ADDITIONAL INFORMATION REQUIRED

The LA must be as complete as possible using information that is reasonably available at the time of submission. If information pertaining to this chapter cannot be supplied at the time of submission, explain why the information was not reasonably available. This information, including schedule of completion, will be included in Section 11.13. The NRC will check for the completion of such items before issuing the license to receive and possess radioactive waste. Authors must not commit to obtain information beyond that needed for the determination of reasonable assurance by the NRC.

8.8 CONCLUSIONS

An executive summary of the conclusions is given in Section 8. Based on the information in Chapter 8, provide a more detailed discussion of the conclusions and relate these conclusions to the uncertainties in and sensitivities of the major data sources and modeling assumptions.

8.8.1 Ten-Thousand-Year Dose Rates

Summarize the results of the nominal and the disturbed scenario cases and present a comparison against the regulatory requirements specified in proposed 10 CFR 63.113(b) (64 FR 8640).

8.8.2 Human Intrusion Dose Rates

Summarize the results of the evaluation of the ability of the potential geologic repository to continue to isolate waste from the accessible environment over the long term in the event of limited human intrusion into the EBS. Explain that no quantitative regulatory limit applies to the results. The results will provide a qualitative indicator of the ability of the potential geologic repository to continue to perform acceptably following human intrusion (Interim Guidance Section 113(d) [Dyer and Horton 1999]).

8.9 REFERENCES

The following references were used to develop this chapter of the technical guidance document. For the LA, this section will contain the references used to develop this chapter of the LA.

References Cited

DOE (U.S. Department of Energy) 1998. *Quality Assurance Requirements and Description.* DOE/RW-0333P, Rev. 8. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.19980601.0022.

Dyer, J.R. and Horton, D.G. 1999. "Interim Guidance Pending Issuance of New Nuclear Regulatory Commission (NRC) Regulations for Yucca Mountain, Nevada." Letter from J.R. Dyer (DOE/YMSCO) to D.R. Wilkins (CRWMS M&O), June 18, 1999, OL&RC:AVG:1435, with enclosure, "Interim Guidance Pending Issuance of New NRC Regulations for Yucca Mountain." ACC: MOL.19990712.0039.

Fabryka-Martin, J.T.; Turin, H.J.; Wolfsberg, A.V.; Brenner, D.; Dixon, P.R.; and Musgrave, J.A. 1996. *Summary Report of Chlorine-36 Studies.* Draft. LA-CST-TIP-96-003. Los Alamos, New Mexico: Los Alamos National Laboratory. ACC: MOL.19970103.0037.

NRC (U.S. Nuclear Regulatory Commission) 1996. *Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program.* NUREG-1563. Washington, D.C.: U.S. Nuclear Regulatory Commission. TIC: 226832.

NRC 1998a. *Issue Resolution Status Report Key Technical Issue: Container Life and Source Term.* Rev. 1. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.19990105.0081.

NRC 1998b. *Issue Resolution Status Report Key Technical Issue: Total System Performance Assessment and Integration.* Rev. 1. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.19990105.0083.

NRC 1998c. *Issue Resolution Status Report Key Technical Issue: Repository Design and Thermal-Mechanical Effects.* Rev. 1. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.19981130.0219.

NRC 1998d. *Issue Resolution Status Report Key Technical Issue: Radionuclide Transport.* Rev. 0. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.19990318.0081.

NRC 1998e. *Issue Resolution Status Report Key Technical Issue: Thermal Effects on Flow.* Rev. 1. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.19990317.0357.

NRC 1998f. *Issue Resolution Status Report Key Technical Issue: Evolution of the Near-Field Environment.* Rev. 1. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.19981106.0144.

NRC 1998g. *Issue Resolution Status Report Key Technical Activity: Igneous Activity.* Rev. 1. Washington D.C.: U. S. Nuclear Regulatory Commission. ACC: MOL.19981014.0058.

NRC 1998h. *Issue Resolution Status Report Key Technical Activity: Unsaturated and Saturated Flow Under Isothermal Conditions*. Rev. 1. Washington D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.19990105.0142.

Yang, I.C.; Yu, P.; Rattray, G.W.; and Thorstenson, D.C. 1997. *Hydrochemical Investigations and Geochemical Modeling in Characterizing the Unsaturated Zone at Yucca Mountain, Nevada*. Draft. Milestone Report 3GUH607M. Denver, Colorado: U.S. Geological Survey. ACC: MOL.19970415.0393.

Codes, Standards, and Regulations

64 FR (Federal Register) 8640. Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, Nevada. Proposed rule 10 CFR 63. Readily available.

CHAPTER 9. RADIOACTIVE WASTE MANAGEMENT

State that this chapter of the License Application (LA) provides a description of the low-level radioactive waste (LLW) management systems for the site-generated wastes at the repository. The authors will produce a chapter that will demonstrate, with reasonable assurance to the U.S. Nuclear Regulatory Commission (NRC), that the site-generated LLW can be safely managed and the systems safely operated. Authors preparing the LA shall use this document for guidance and must read the Introduction and Appendix B before writing their respective sections.

Proposed 10 CFR 63.24(a) (64 FR 8640) recognizes that all information may not be available at the time of docketing the LA. The guidance in this chapter differentiates the information required at the time of construction authorization (CA) from that required at the time of license to receive and possess high-level radioactive waste and spent nuclear fuel. In the following sections, information identified as "Information Required at the Time of Construction Authorization" will be provided at the time the LA is submitted for docketing. In the future, information that was provided at the time of CA for docketing will be updated, and information developed since the time of docketing the LA will be added as noted in this guidance. The additional information that will be provided in the future is identified as "Information Required at the Time of Update to the License Application to Receive and Possess."

Because the LLW systems design has licensing precedent, the level of detail required in Chapter 9 is expected to be less than is required for the structures, systems, and components (SSCs) that do not have licensing precedent. The minimum required level of design detail for the LLW SSCs is provided in Section 9.1.

9. PURPOSE AND SUMMARY

Provide an introductory discussion outlining the purpose and organization of Chapter 9 of the LA. Provide a list of the functional objectives of the overall LLW management program. Introduce the individual LLW management systems.

Provide a summary and conclusions regarding how the LLW management systems design methodology and specific designs support safe operation of the repository and the repository safety case.

9.1 LOW-LEVEL RADIOACTIVE WASTE MANAGEMENT SYSTEMS OVERVIEW

The purpose of Section 9.1 of the LA is to provide the reader a general overview of the LLW management systems as a whole.

Information Required at the Time of Construction Authorization–Provide a general description of the LLW management systems for site-generated wastes at the geologic repository operations area (GROA). State that the repository design does not include a gaseous LLW management system and state why one is not necessary (provide reference to Section 9.6 for source term information). Refer to Section 4.4.7 for a discussion of high-efficiency particulate air (HEPA) filtration for airborne radioactive waste. State that the design of the liquid LLW

management system precludes liquid effluent to the unrestricted environment, and provide a brief explanation.

Identify the major operations performed within each of the LLW management systems. Provide a brief overview of the site-generated waste streams to be processed. Describe the flow paths, relationships, and interfaces between the LLW management systems and between other portions of the GROA. Provide the location of the LLW management systems at the GROA. Provide drawings or figures to identify the LLW management systems. Provide key descriptive parameters for the LLW management systems, such as the capacities of the systems, construction type, number of employees, site area, overall floor space, and site location.

State that the repository is designed to withstand natural phenomena; provide reference to Chapter 7 for the specific natural phenomena considered and evaluated for design basis events (DBEs). State that DBE analyses performed to date do not identify any DBEs for the LLW management systems. Provide references to these analyses.

Provide a reference to Section 4.1 for a description of the U.S. Department of Energy and Civilian Radioactive Waste Management System (CRWMS) Management and Operating Contractor (M&O) documents that direct and control the repository design such as the generic requirements documents, the system description documents (SDDs), and system and subsystem requirements documents. State that the LLW management system designs also meet any other legal requirements of federal, state, and local agencies applicable to the mined geologic disposal of spent nuclear fuel and high-level radioactive waste.

Briefly discuss the integrated safety analysis (ISA) performed to ensure the preclosure performance objectives are satisfied. Provide a reference to Chapter 7 of the LA for a detailed discussion of the ISA. Briefly discuss how the ISA results are reflected in this chapter.

Information Required at the Time of Update to the License Application to Receive and Possess–Evaluate the statements regarding exclusion of a gaseous waste management system, preclusion of liquid effluent release, and no identified Category 1 or 2 DBEs associated with the LLW management system. Revise text as necessary if such conditions have changed with the evolution of design and analyses.

9.1.1 Level of Design Detail

This section provides the guidance for the required level of detail that must be provided for the SSCs that are discussed in Chapter 9. There will be no Section 9.1.1 in the LA.

Provide the level of design detail given below for each of the LLW management SSCs, as applicable.

Quality Level (QL) 1, 2, and 3 SSCs and non-safety SSCs are required to be described in the LA per the *Level of Design Detail Necessary for the License Application for Construction Authorization* (CRWMS M&O 1999). Provide a reference to Section 2.4 for more information related to QL classifications.

For any SSC, the level of design detail required depends on the following (CRWMS M&O 1999):

- Importance to protection of public health and safety
- Importance to protection of worker health and safety
- Need to demonstrate compliance with regulatory requirements
- Need to support submittal of a docketable LA.

The QL information following this paragraph lists the specific information to be included in the description of SSCs that fall into one of the three QL classifications. If a specific information item is not relevant for a given SSC, that information item need not be addressed for that SSC in the LA.

9.1.1.1 Quality Level 1 Structures, Systems, and Components

The following information, as applicable, must be provided for each QL 1 SSC.

Regulatory Bases–These are primarily the proposed 10 CFR 63 (64 FR 8640) requirements.

- Identify the SSCs relied upon to limit or prevent potential accidents or mitigate their consequences; refer to Chapter 7 for details on the analysis that identified the SSC as such. Identify the specific DBEs. Describe the design features incorporated into the SSC and describe the function of the SSCs, including controls that are relied upon to limit, prevent, or mitigate the consequences of DBEs. Use drawings such as piping and instrument diagrams, electrical one-line diagrams, general arrangement drawings, and handling drawings as necessary to present the information. Identify measures taken to ensure the availability of identified safety systems (Interim Guidance Section 112(e) [Dyer and Horton 1999]). Discuss the design considerations that prevent releases of radioactive materials that could result in a dose of 0.25 mSv (25 mrem) to an individual member of the public at the boundary. Where appropriate, identify the sequence of events and how the system responds to the event (Interim Guidance Section 111(a)(2) [Dyer and Horton 1999]; proposed 10 CFR 63.111(b)(1) [64 FR 8640]; Interim Guidance Section 111(b)(2) [Dyer and Horton 1999]).

- Identify the SSCs that limit and control radiation exposures and radiation levels in restricted and unrestricted areas and the release of radioactive materials to unrestricted areas, and address the limits of 10 CFR 20.1201, 10 CFR 20.1301 and the as low as is reasonably achievable (ALARA) provisions of 10 CFR 20.1101(b) and 10 CFR 20.1101(d). The limits are as follows (proposed 10 CFR 63.111(a)(1) [64 FR 8640]; Interim Guidance Section 111(a)(2) [Dyer and Horton 1999]; proposed 10 CFR 63.111(b)(1) [64 FR 8640]):

  - No worker shall receive the more limiting total effective dose equivalent (TEDE) of 0.05 Sv (5 rem), or the sum of deep-dose equivalent and committed dose equivalent, to any individual organ or tissue (other than the lens of the eye) of 0.50 Sv (50 rem) annually. The annual dose equivalent to lens of the eye shall not exceed 0.15 Sv

(15 rem). The annual shallow dose equivalent to the skin or any extremity shall not exceed 0.50 Sv (50 rem).

- No individual member of the public shall receive a TEDE in excess of 1 mSv (0.1 rem) annually, and the dose in any unrestricted area from external sources does not exceed 0.02 mSv (0.002 rem) per hour.

- Identify ALARA design considerations for facility features that limit and control occupational dose and dose to members of the public as required by 10 CFR 20.1101(b) and 10 CFR 20.1101(d) (proposed 10 CFR 63.111(a)(1) [64 FR 8640]).

- Describe the design considerations for systems that monitor and control effluents. Describe design considerations for facility features and systems that control and monitor radiation levels to limit occupational radiation exposure (proposed 10 CFR 63.21(c)(14) and proposed 10 CFR 63.111(a)(1) [64 FR 8640]).

License Specifications

- **Information Required at the Time of Construction Authorization**–When discussing equipment or parameters credited in the safety analysis for mitigating the consequences of a Category 1 or 2 DBE, note that the item is being considered as a subject for a license specification in accordance with proposed 10 CFR 63.21(c)(13) (64 FR 8640). Include supporting information to demonstrate how and why the item is credited in the safety analysis for mitigating the consequences of a Category 1 or 2 DBE. Provide a reference to Chapter 11 for an identification of the license specifications and summary of their justification for being considered as a license specification (proposed 10 CFR 63.21(c)(13) [64 FR 8640]).

- **Information Required at the Time of Update to the License Application to Receive and Possess**–Identify equipment and parameters used to mitigate the consequences of a Category 1 or 2 DBE as being addressed in the license specifications. Provide a reference to Chapter 11 for a discussion of the license specification development process (proposed 10 CFR 63.21(c)(13) [64 FR 8640]).

Applicable Codes and Standards–List the codes and standards (including guidance documents and technical positions) required in the design and construction of the LLW management system. The codes and standards should be listed on a structure or system level. The codes and standards listed must be the same as those found in corresponding SDDs (proposed 10 CFR 63.21(c)(3) [64 FR 8640]).

Design Criteria–A design criterion is a standard or rule against which a design can be judged. The NRC defines principal design criteria in 10 CFR 50, Appendix A, as criteria that establish the necessary design, fabrication, construction, and performance requirements for SSCs important to safety (CRWMS M&O 1999). Examples include:

- The lifetime of the waste package (WP) shall be long enough to contain the waste throughout the thermal period.

- Provisions shall be made so that, if there is a loss of the primary electric power source or circuit, reliable and timely emergency power can be provided to instruments, utility service systems, and operating systems, including alarm systems, important to safety.

- The facility design shall comply with the ALARA criteria of 10 CFR 20.

- See the example table below.

Summary of Centralized Interim Storage Facility Principal Design Criteria

Design Parameter	Design Criteria	Condition	Applicable Codes, Standards, & Bases
Seismic (Ground Motion)	Design response spectra anchored at horizontal acceleration of 0.75g	Accident	N/A
Seismic (Surface Faulting)	No surface faulting	Accident	N/A

Source: DOE 1998

Include the design criteria for the LLW management system, preferably in tabular format. The design criteria are found in the appropriate SDD.

Design Bases–Design bases refer to the information that identifies the specific functions to be performed by an SSC of a facility and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be restraints derived from generally accepted "state-of-the-art" practices for achieving functional goals or requirements derived from analysis (based on calculation or experiments) of the effects of a postulated event under which an SSC must meet its functional goals. The values for controlling parameters for external events include (CRWMS M&O 1999):

- Estimates of severe natural events, to be used for deriving design bases, that will be based on consideration of historical data on the associated parameters, physical data, or analysis of upper limits of the physical processes involved.

- Estimates of severe external human-induced events, to be used for deriving design bases that will be based on analysis of human activity in the region, taking into account the site characteristics and the risks associated with the event.

Include the design bases for the LLW management system SSCs. The project design bases are the combination of the system functions and the performance parameters found in the SDDs.

General Description–Provide a general description of the SSCs, equipment, and process activities. Base this description on the information contained in the related SDDs. Include in this description the information required to support the safety analysis of the system or that which can be readily derived from it. Provide a summary of the system functions, operations, the system design, concept of operations, and a description of system interfaces. Include a discussion on any special construction or fabrication techniques, unique testing programs, or special design and analysis procedures used for the SSCs, as applicable. Include diagrams to show concepts or ideas as needed to support the text (proposed 10 CFR 63.112(a) [64 FR 8640]).

Discussion of the materials of construction (including general arrangement and approximate dimensions) for the GROA (proposed 10 CFR 63.21(c)(3) [64 FR 8640]):

- **Information Required at the Time of Construction Authorization**–Include only the principal materials used in the design of SSCs that either prevent or mitigate a Category 1 or 2 DBE or are required for worker safety. Examples of the principal materials are structural descriptions such as steel beam construction or reinforced concrete, systems components such as stainless steel piping, HEPA filters with charcoal absorbers, or leaded glass used for shielding. The materials of construction should be discussed on an SSC level as they are described in the various sections.

- **Information Required at the Time of Update to the License Application to Receive and Possess**–In addition to information provided at the time of CA, present a detailed description of the materials of construction. This should be done by component or subsystem as each is discussed in the description of the specific SSCs. The information should include the compatibility of the material with its environment during normal operations or post-accident situations, whichever is the most limiting as it relates to the service that the component provides. Three examples are provided below:

  – HEPA filters

    HEPA filter elements are of pleated fiberglass with aluminum separator design, measure 24 x 24 x 11.5 inches, and are each capable of handling a nominal flowrate of 1000 ft^3/min. The filter medium is cased in stainless steel, has face guards on both sides, and is water- and fire-resistant.

  – Cooling coils

    The cooling coils are of nonferrous construction with aluminum fins mechanically bonded to seamless copper tubing. Coils are arranged for counter-flow operation using chilled water. The tube bundle is enclosed in a steel frame.

  – Low total dissolved solids holdup tank (T-01 C)

Quantity/unit	= 1
Capacity (each)	= 30,000 gal
Design pressure and temperature	= Atmospheric pressure and 150°F
Operating pressure and temperature	= Atmospheric pressure and 80°F
Material	= 304 stainless steel

Research and Development

- **Information Required at the Time of Construction Authorization**–Identify SSCs that require research and development (R&D) to confirm the adequacy of design. Provide available information that describes the type of R&D required and the reason the additional information is needed. Provide a reference to Chapter 11 for a description of

the R&D program and the proposed schedule (proposed 10 CFR 63.21(c)(21) [64 FR 8640]).

- **Information Required at the Time of Update to the License Application to Receive and Possess**–Provide the results from the required R&D identified in the LA at the time of CA for the various SSCs. Discuss the reason the additional information was originally needed (proposed 10 CFR 63.21(c)(21) [64 FR 8640]).

Decontamination

- **Information Required at the Time of Construction Authorization**–Provide a reference to Section 4.7 for a discussion on decommissioning.

- **Information Required at the Time of Update to the License Application to Receive and Possess**–Provide the specific decontamination design considerations as part of the SSC discussion (proposed 10 CFR 63.21(c)(18) [64 FR 8640]). Ensure compatibility with the Section 4.7 discussion.

Drawings and Diagrams–The drawings for QL 1 SSCs included in the LA are those that show information needed to support the safety case; the other specified drawings are not required (proposed 10 CFR 63.112(a) [64 FR 8640]).

- Piping and instrument diagrams
- Electrical one-line diagrams
- General arrangement drawings
- Handling diagrams.

9.1.1.2 Quality Level 2 Structures, Systems, and Components

The following information, as applicable, must be provided for each QL 2 SSC.

Regulatory Bases–These are primarily the proposed 10 CFR 63 (64 FR 8640) requirements.

- Identify the SSCs relied upon to limit or prevent potential accidents or mitigate their consequences; refer to Chapter 7 for details on the analysis that identified the SSC as such. Identify the specific DBEs. Identify and describe the design features incorporated into the SSC and describe the function of the SSCs, including controls that are relied upon to limit, prevent, or mitigate the consequences of DBEs. Include identification of measures taken to ensure the availability of identified safety systems (Interim Guidance Section 112(e) [Dyer and Horton 1999]). Discuss the design considerations that prevent releases of radioactive materials that could result in a dose of 0.25 mSv (25 mrem) to an individual member of the public at the boundary. Where appropriate, identify the sequence of events and how the system responds to the event (Interim Guidance Section 111(a)(2) [Dyer and Horton 1999]; proposed 10 CFR 63.111(b)(1) [64 FR 8640]; Interim Guidance Section 111(b)(2) [Dyer and Horton 1999]).

- Identify the SSCs that limit and control radiation exposures and radiation levels in restricted and unrestricted areas and the release of radioactive materials to unrestricted

areas, and address the limits of 10 CFR 20.1201, 10 CFR 20.1301 and the ALARA provisions of 10 CFR 20.1101(b) and 10 CFR 20.1101(d). The limits are as follows (proposed 10 CFR 63.111(a)(1) [64 FR 8640]; Interim Guidance Section 111(a)(2) [Dyer and Horton 1999]; proposed 10 CFR 63.111(b)(1) [64 FR 8640]):

- No worker shall receive the more limiting TEDE of 0.05 Sv (5 rem), or the sum of deep-dose equivalent and committed dose equivalent, to any individual organ or tissue (other than the lens of the eye) of 0.50 Sv (50 rem) annually. The annual dose equivalent to lens of the eye shall not exceed 0.15 Sv (15 rem). The annual shallow dose equivalent to the skin or any extremity shall not exceed 0.50 Sv (50 rem).

- No individual member of the public shall receive a TEDE in excess of 1 mSv (0.1 rem) annually, and the dose in any unrestricted area from external sources does not exceed 0.02 mSv (0.002 rem) per hour.

- Identify ALARA design considerations for facility features that limit and control occupational dose and dose to members of the public as required by 10 CFR 20.1101(b) and 10 CFR 20.1101(d) (proposed 10 CFR 63.111(a)(1) [64 FR 8640]).

- Describe the design considerations for systems that monitor and control effluents. Describe design considerations for facility features and systems that control and monitor radiation levels to limit occupational radiation exposure (proposed 10 CFR 63.21(c)(14) and proposed 10 CFR 63.111(a)(1) [64 FR 8640]).

License Specifications

- **Information Required at the Time of Construction Authorization**–When discussing equipment or parameters credited in the safety analysis for mitigating the consequences of a Category 1 or 2 DBE, note that the item is being considered as a subject for a license specification in accordance with proposed 10 CFR 63.21(c)(13) (64 FR 8640). Include the supporting information to demonstrate how and why the item is credited in the safety analysis for mitigating the consequences of a Category 1 or 2 DBE. Refer to Chapter 11 for an identification of the license specifications and summary of their justification for being considered as a license specification (proposed 10 CFR 63.21(c)(13) [64 FR 8640]).

- **Information Required at the Time of Update to the License Application to Receive and Possess**–Identify equipment and parameters used to mitigate the consequences of a Category 1 or 2 DBE as being addressed in the license specifications. Refer to Chapter 11 for a discussion of the license specification development process (proposed 10 CFR 63.21(c)(13) [64 FR 8640]).

Applicable Codes and Standards–List the codes and standards (including guidance documents and technical positions) required in the design and construction of the LLW management system. The codes and standards should be listed on a structure or system level. The codes and standards listed must be the same as those found in corresponding SDDs (proposed 10 CFR 63.21(c)(3) [64 FR 8640]).

Design Criteria–A design criterion is a standard or rule against which a design can be judged. The NRC defines principal design criteria in 10 CFR 50, Appendix A, as criteria that establish the necessary design, fabrication, construction, and performance requirements for SSCs important to safety (CRWMS M&O 1999). Examples include:

- The lifetime of the WP shall be long enough to contain the waste throughout the thermal period.
- Provisions shall be made so that, if there is a loss of the primary electric power source or circuit, reliable and timely emergency power can be provided to instruments, utility service systems, and operating systems, including alarm systems, important to safety.
- The facility design shall comply with the ALARA criteria of 10 CFR 20.
- See the example table in Section 9.1.1.1.

Include the design criteria for the LLW management system, preferably in tabular format. The design criteria are found in the appropriate SDD.

General Description–A general description will be made of the SSCs, equipment, and process activities. Base this description on the information contained in the related SDDs. Include in this description the information required to support the safety analysis of the system or that which can be readily derived from it. Provide a summary of the system functions, operations, the system design, concept of operations, and a description of system interfaces. Include a discussion on any special construction or fabrication techniques, unique testing programs, or special design and analysis procedures used for the SSCs, as applicable. Include diagrams to show concepts or ideas as needed to support the text (proposed 10 CFR 63.112(a) [64 FR 8640]).

Discussion of the materials of construction (including general arrangement and approximate dimensions) for the GROA (proposed 10 CFR 63.21(c)(3) [64 FR 8640]):

- **Information Required at the Time of Construction Authorization**–Include only the principal materials used in the design of SSCs that either prevent or mitigate a Category 1 or 2 DBE or are required for worker safety. Examples of the principal materials are structural descriptions such as steel beam construction or reinforced concrete, systems components such as stainless steel piping, HEPA filters with charcoal absorbers, or leaded glass used for shielding. The materials of construction should be discussed on an SSC level as they are described in the various sections.
- **Information Required at the Time of Update to the License Application to Receive and Possess**–In addition to information provided at the time of CA, a detailed description of the materials of construction must be presented. This should be done by component as each is discussed in the description of the specific SSCs. The information should include the compatibility of the material with its environment during normal operations or post-accident situations, whichever is the most limiting and as it relates to the service that the component provides. See the examples provided under QL 1 SSCs.

Research and Development

- **Information Required at the Time of Construction Application**–Identify SSCs that require R&D to confirm the adequacy of design. Provide available information that describes the type of R&D required and the reason the additional information is needed. Provide a reference to Chapter 11 for a description of the R&D program and the proposed schedule (proposed 10 CFR 63.21(c)(21) [64 FR 8640]).

- **Information Required at the Time of Update to the License Application to Receive and Possess**–Provide the results from the required R&D identified in the LA at the time of CA for the various SSCs. Discuss the reason the additional information was originally needed (proposed 10 CFR 63.21(c)(21) [64 FR 8640]).

Decontamination

- **Information Required at the Time of Construction Authorization**–Provide a reference to Section 4.7 for a discussion on decommissioning.

- **Information Required at the Time of Update to the License Application to Receive and Possess**–Provide the specific decontamination design considerations as part of the SSC discussion (proposed 10 CFR 63.21(c)(18) [64 FR 8640]). Ensure compatibility with the Section 4.7 discussion.

9.1.1.3 Quality Level 3 Structures, Systems, and Components

The information, as applicable, that must be provided for each QL 3 SSC includes the following.

Regulatory Bases–These are primarily the proposed 10 CFR 63 (64 FR 8640) requirements.

- Identify the SSCs relied upon to limit or prevent potential accidents or mitigate their consequences; refer to Chapter 7 for details on the analysis that identified the SSC as such. Identify and describe the function of the SSCs, including controls that are relied upon to limit, prevent, or mitigate the consequences of DBEs. Include identification of measures taken to ensure the availability of identified safety systems (Interim Guidance Section 112(e) [Dyer and Horton 1999]). Discuss the design considerations that prevent releases of radioactive materials that could result in a dose of 0.25 mSv (25 mrem) to an individual member of the public at the boundary. Where appropriate, identify the sequence of events and how the system responds to the event (Interim Guidance Section 111(a)(2) [Dyer and Horton 1999]; proposed 10 CFR 63.111(b)(1) [64 FR 8640]; Interim Guidance Section 111(b)(2) [Dyer and Horton 1999]).

- Identify the SSCs that limit and control radiation exposures and radiation levels in restricted and unrestricted areas and the release of radioactive materials to unrestricted areas, and address the limits of 10 CFR 20.1201, 10 CFR 20.1301 and the ALARA provisions of 10 CFR 20.1101(b) and 10 CFR 20.1101(d). The limits are as follows (proposed 10 CFR 63.111(a)(1) [64 FR 8640]; Interim Guidance Section 111(a)(2) [Dyer and Horton 1999]; proposed 10 CFR 63.111(b)(1) [64 FR 8640]):

- No worker shall receive the more limiting TEDE of 0.05 Sv (5 rem), or the sum of deep-dose equivalent and committed dose equivalent, to any individual organ or tissue (other than the lens of the eye) of 0.50 Sv (50 rem) annually. The annual dose equivalent to lens of the eye shall not exceed 0.15 Sv (15 rem). The annual shallow dose equivalent to the skin or any extremity shall not exceed 0.50 Sv (50 rem).

- No individual member of the public shall receive a TEDE in excess of 1 mSv (0.1 rem) annually, and the dose in any unrestricted area from external sources does not exceed 0.02 mSv (0.002 rem) per hour.

- Identify ALARA design considerations for facility features that limit and control occupational dose and dose to members of the public as required by 10 CFR 20.1101(b) and 10 CFR 20.1101(d) (proposed 10 CFR 63.111(a)(1) [64 FR 8640]).

- Describe the design considerations for systems that monitor and control effluents. Describe design considerations for facility features and systems that control and monitor radiation levels to limit occupational radiation exposure (proposed 10 CFR 63.21(c)(14) and proposed 10 CFR 63.111(a)(1) [64 FR 8640]).

License Specifications

- **Information Required at the Time of Construction Authorization**–When discussing equipment or parameters credited in the safety analysis for mitigating the consequences of a Category 1 or 2 DBE, note that the item is being considered as a subject for a license specification in accordance with proposed 10 CFR 63.21(c)(13) (64 FR 8640). Include supporting information to demonstrate how and why the item is credited in the safety analysis for mitigating the consequences of a Category 1 or 2 DBE. Refer to Chapter 11 for an identification of the license specifications and summary of their justification for being considered as a license specification (proposed 10 CFR 63.21(c)(13) [64 FR 8640]).

- **Information Required at the Time of Update to the License Application to Receive and Possess**–Identify equipment and parameters used to mitigate the consequences of a Category 1 or 2 DBE as being addressed in the license specifications. Refer to Chapter 11 for a discussion of the license specification development process (proposed 10 CFR 63.21(c)(13) [64 FR 8640]).

Design Criteria–A design criterion is a standard or rule against which a design can be judged. The NRC defines principal design criteria in 10 CFR 50, Appendix A, as criteria that establish the necessary design, fabrication, construction, and performance requirements for SSCs important to safety (CRWMS M&O 1999). Examples include:

- The lifetime of the WP shall be long enough to contain the waste throughout the thermal period.
- Provisions shall be made so that, if there is a loss of the primary electric power source or circuit, reliable and timely emergency power can be provided to instruments, utility service systems, and operating systems, including alarm systems, important to safety.
- The facility design shall comply with the ALARA criteria of 10 CFR 20.
- See the example table in Section 9.1.1.1.

Include the design criteria for the LLW management system, preferably in tabular format. The design criteria are found in the appropriate SDD.

General Description–A general description will be made of the SSCs, equipment, and process activities. Base this description on the information contained in the related SDDs. Include in this description the information required to support the safety analysis of the system or that which can be readily derived from it. Provide a summary of the system functions, operations, the system design, concept of operations, and a description of system interfaces. Include a discussion on any special construction or fabrication techniques, unique testing programs, or special design and analysis procedures used for the SSCs, as applicable. Include diagrams to show concepts or ideas as needed to support the text (proposed 10 CFR 63.112(a) [64 FR 8640]).

Discussion of the materials of construction (including general arrangement and approximate dimensions) for the GROA (proposed 10 CFR 63.21(c)(3) [64 FR 8640]).

- **Information Required at the Time of Construction Authorization**–Include only the principal materials used in the design of SSCs that either prevent or mitigate a Category 1 or 2 DBE or are required for worker safety. Examples of the principal materials are structural descriptions such as steel beam construction or reinforced concrete, systems components such as stainless steel piping, HEPA filters with charcoal absorbers, or leaded glass used for shielding. The materials of construction should be discussed on an SSC level as they are described in the various sections.
- **Information Required at the Time of Update to the License Application to Receive and Possess**–In addition to information provided at the time of CA, a detailed description of the materials of construction must be presented. This should be done by component as each is discussed in the description of the specific SSCs. The information should include the compatibility of the material with its environment during normal operations or post-accident situations, whichever is the most limiting and as it relates to the service that the component provides. See the examples provided under QL 1 SSCs.

Research and Development

- **Information Required at the Time of Construction Authorization**–Identify SSCs that require R&D to confirm the adequacy of design. Provide available information that describes the type of R&D required and the reason the additional information is needed. Provide a reference to Chapter 11 for a description of the R&D program and the proposed schedule (proposed 10 CFR 63.21(c)(21) [64 FR 8640]).

- **Information Required at the Time of Update to the License Authorization to Receive and Possess**–Provide the results from the required R&D identified in the LA at the time of CA for the various SSCs. Discuss the reason the additional information was originally needed (proposed 10 CFR 63.21(c)(21) [64 FR 8640]).

Decontamination

- **Information Required at the Time of Construction Authorization**–Provide a reference to Section 4.7 for the discussion on decommissioning.

- **Information Required at the Time of Update to the License Application to Receive and Possess**–Provide the specific decontamination design considerations as part of the SSC discussion (proposed 10 CFR 63.21(c)(18) [64 FR 8640]). Ensure compatibility with the Section 4.7 discussion.

9.1.1.4 Non-Safety Systems, Structures, and Components

The information that must be provided for any non-safety SSCs discussed in this chapter includes:

Regulatory Bases–These are primarily the proposed 10 CFR 63 (64 FR 8640) requirements.

General Description–Provide a general description only to the extent that is sufficient to demonstrate the non-safety classification, which should be based on information contained in Section 1 of the related SDDs. Other diagrams may be included to show concepts or ideas to support the text to the extent needed to demonstrate the non-safety classification (proposed 10 CFR 63.112(a) [64 FR 8640]).

Decontamination

- **Information Required at the Time of Construction Authorization**–Provide a reference to Section 4.7 for the discussion on decommissioning.

- **Information Required at the Time of Update to the License Application to Receive and Possess**–Provide the specific decontamination design considerations as part of the SSC discussion (proposed 10 CFR 63.21(c)(18) [64 FR 8640]). Ensure compatibility with the Section 4.7 discussion.

9.2 LIQUID LOW-LEVEL RADIOACTIVE WASTE MANAGEMENT SYSTEM

This section discusses the liquid LLW management system adequately to demonstrate reasonable assurance to the NRC that the safety of the public, worker, and environment can be protected.

Provide the required level of design detail for this system and equipment, as outlined in Sections 9.1 through 9.1.1.4, taking into consideration the following information.

Provide a general, overall description of the liquid LLW management system and the functional objectives of the system. State that no liquid effluents are planned or expected as part of normal, off-normal, or accident conditions. Discuss interfaces with other LLW management systems. Identify the QL classification for each SSC discussed.

9.3 SOLID LOW-LEVEL WASTE MANAGEMENT SYSTEM

This section discusses the solid LLW management system adequately to demonstrate reasonable assurance to the NRC that the safety of the public, worker, and environment can be protected.

Provide the required level of design detail for this system and equipment, as outlined in Sections 9.1 through 9.1.1.4, taking into consideration the following information.

Provide a general, overall description of the solid LLW management system and the functional objectives of the system. State that this system is the primary means of processing non-recyclable liquid wastes and solid wastes for offsite disposal. Discuss interfaces with other LLW management systems. Identify the QL classification for each SSC discussed.

9.4 MIXED WASTE MANAGEMENT SYSTEM

This section discusses the mixed waste management system adequately to demonstrate assurance to the NRC that the safety of the public, worker, and environment can be protected.

Provide the required level of design detail for this system and equipment, as outlined in Sections 9.1 through 9.1.1.4, taking into consideration the following information.

Provide a general, overall description of the mixed waste management system and the functional objectives of the system. State that design, administrative, and operational controls will prevent, or greatly lessen, the generation of mixed waste, but that provisions will be in place for handling, staging, storing, and disposing of such waste if necessary. Discuss interfaces with other LLW management systems. Identify the QL classification for each SSC discussed.

9.5 PROCESS RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

This section discusses the process radiological monitoring and sampling system for the LLW management systems adequately to demonstrate assurance to the NRC that the safety of the public, worker, and environment can be protected.

Provide the required level of design detail for this system and equipment, as outlined in Sections 9.1 through 9.1.1.4, taking into consideration the following information.

Provide a general, overall description of the process radiological monitoring and sampling system for the radioactive waste management systems and the functional objectives of the system. State that no liquid effluents are planned or expected as part of normal, off-normal, or accident conditions. Reference Chapter 10 for discussion of the radiological environmental monitoring and sampling program. Discuss interfaces with the LLW management systems. Identify the QL classification for each SSC discussed.

9.6 SOURCE TERMS

This section provides guidance to identify the site-generated radioactive waste, the predicted characteristics and amounts of such waste, and potential release pathways. This section of the TGD provides the guidance necessary to demonstrate reasonable assurance to the NRC that the safety of the public, worker, and environment can be protected.

State that the purpose of this section is to characterize the radioactive waste to be generated during repository operations, those released within the GROA, and those released to the environment, if any. This characterization will support the dose assessments described in Chapters 7 and 10 of the LA.

Describe the anticipated characteristics of the radioactive waste (including radioactive effluents, if any) associated with normal operations at the repository, anticipated operational occurrences, and accident situations. Identify potential release pathways to the environment. State, however, that with proper design and operational limits, effluents are not anticipated.

Provide a summary that identifies each type of waste (liquid, solid, and gaseous LLW, as well as mixed waste). Present the data by waste type, and present it in subsections (e.g., 9.6.1 Liquid Waste, 9.6.2 Solid Waste, etc.) if appropriate. Identify the anticipated amount of site-generated waste per metric ton (or other unit) and of waste handled and stored per unit of time. Include bases or assumptions to substantiate the discussions. Identify mathematical models used to calculate source terms where applicable.

9.7 REFERENCES

The following references were used to develop this chapter of the TGD. For the LA, this section will contain the references used to develop this chapter of the LA.

References Cited

CRWMS (Civilian Radioactive Waste Management System) M&O (Management and Operating Contractor) 1999. *Level of Design Detail Necessary for the License Application for Construction Authorization.* B00000000-01717-1710-00003 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990708.0065.

DOE (U.S. Department of Energy) 1998. *Centralized Interim Storage Facility Topical Safety Analysis Report* . BA0000000-01717-5700-00017 REV 01. Two volumes. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.19990212.0117.

Dyer, J.R. and Horton, D.G. 1999. "Interim Guidance Pending Issuance of New Nuclear Regulatory Commission (NRC) Regulations for Yucca Mountain, Nevada." Letter from J.R. Dyer (DOE/YMSCO) to D.R. Wilkins (CRWMS M&O), June 18, 1999, OL&RC:AVG:1435, with enclosure, "Interim Guidance Pending Issuance of New NRC Regulations for Yucca Mountain." ACC: MOL.19990712.0039.

Codes, Standards, and Regulations

10 CFR (Code of Federal Regulations) 20. Energy: Standards for Protection Against Radiation. Readily available.

10 CFR 50. Energy: Domestic Licensing of Production and Utilization Facilities. Readily available.

64 FR (Federal Register) 8640. Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, Nevada. Proposed rule 10 CFR 63. Readily available.

CHAPTER 10. RADIATION PROTECTION

This chapter of the License Application (LA) provides a description of the radiation protection programs for the potential monitored geologic repository (MGR) at Yucca Mountain. These programs when implemented will ensure management of the identified radioactive material and waste, performance confirmation activities, operation and eventual closure of the repository, subsequent decommissioning of surface facilities, and license termination activities without exceeding occupational and public dose limits (proposed 10 CFR 63.111(a)(1) [64 FR 8640]; Interim Guidance Section 111(a)(2) [Dyer and Horton 1999]; proposed 10 CFR 63.111(b) [64 FR 8640]). In addition to compliance with regulatory limits, occupational doses and doses to the public will be maintained as low as is reasonably achievable (ALARA) (10 CFR 20.1101). The authors will produce a chapter that will enable the U.S. Nuclear Regulatory Commission (NRC) to determine with reasonable assurance that the repository can be safely operated. In other words, the required level of detail is that needed to demonstrate compliance with regulations. Demonstrating compliance involves providing sufficient technical basis to allow the NRC to determine that there is reasonable assurance that the repository can and will be designed, constructed, and operated without unreasonable risk to the health and safety of the public, and to demonstrate that there is reasonable assurance that postclosure performance objectives will be met, consistent with the regulations.

Proposed 10 CFR 63.24(b) (64 FR 8640) recognizes that all information may not be available at the time of docketing the LA. The guidance in this chapter adopts a differentiated approach for providing the information required at construction authorization (CA) as opposed to that required at license to receive and possess high-level radioactive waste (HLW). Information in any section that is identified as "Information Required at the Time of Construction Authorization" will be provided at the time the LA is submitted for docketing. The information at the CA stage will be replaced with additional information developed since docketing the LA as appropriate. The additional information is identified as "Information Required at the Time of Update to the License Application to Receive and Possess."

Authors preparing the LA shall use this document for guidance and must read the Introduction and Appendix B before writing their respective sections. Compliance Program Guidance Packages have been prepared related to the contents of this chapter. Ensure all applicable requirements, as identified in the appropriate compliance program guidance packages, are addressed.

10. PURPOSE AND SUMMARY

This section will present a summary of the radiation protection information presented in this chapter. State the purpose of this chapter of the LA. Describe the organization of the chapter.

10.1 AS LOW AS IS REASONABLY ACHIEVABLE PROGRAM

This section will present an overview of the U.S. Department of Energy (DOE) ALARA considerations for radiation protection. State that the DOE ALARA program implements the requirements of 10 CFR 20.1101 for ALARA, as defined in 10 CFR 20.1003, to ensure that occupational radiation doses and doses to members of the public are ALARA (proposed

10 CFR 63.21(c)(14) [64 FR 8640]). State that the sections that follow identify MGR ALARA considerations.

10.1.1 As Low as Is Reasonably Achievable in Design

This section will describe ALARA policy with respect to designing and constructing the MGR. Describe the applicable organizational responsibilities and related activities to be conducted by managers accountable during design and construction for radiation protection. Describe the policy of maintaining occupational doses and doses to members of the public ALARA. Indicate how the ALARA in design program satisfies the requirements of 10 CFR 20.1101(b); proposed 10 CFR 63.21(c)(14), proposed 10 CFR 63.111(a)(1), and proposed 10 CFR 63.111(b)(1) (64 FR 8640); and Interim Guidance Section 111(a)(2) (Dyer and Horton 1999). Provide reference to the appropriate section in Chapter 7 for discussion of ALARA as it relates to Category 1 design basis event (DBE) consequences. Describe the processes that provide for design review by a competent radiation protection professional. Describe how operational requirements are reflected in the design considerations and radiation protection design features. Provide reference to appropriate sections in Chapters 4, 6, and 9 for design bases, system description, and design evaluation of facility features for ALARA design.

10.1.2 Operational as Low as Is Reasonably Achievable Considerations

This section will discuss ALARA policy and program for assuring that occupational radiation doses and doses to members of the public are ALARA for the operation and eventual closure of the MGR and subsequent decommissioning of surface facilities.

Information Required at the Time of Construction Authorization–State that an operational ALARA program and policies will be implemented. Identify the elements of the planned operational ALARA program. Indicate how the operational ALARA program will satisfy the requirements of 10 CFR 20.1101(b); proposed 10 CFR 63.21(c)(14), proposed 10 CFR 63.111(a)(1), and proposed 10 CFR 63.111(b)(2) (64 FR 8640); and Interim Guidance Section 111(a)(2) (Dyer and Horton 1999).

Information Required at the Time of Update to the License Application to Recieve and Possess–Describe the responsibilities and related activities to be conducted by managers accountable for radiation protection. Describe the policy of maintaining occupational exposures ALARA. Indicate how the program applies to MGR operations and complies with the requirements of 10 CFR 20.1101(b) and 10 CFR 20.1101(d). Provide the criteria for operating procedures and techniques for implementation of ALARA for all systems that contain, collect, store, or transport radioactive liquids, gases, and solids. Describe the process for planning and developing procedures for such radiation exposure-related operations as maintenance, in-service inspections, performance confirmation, and radioactive waste handling in a manner that will ensure that the exposures are ALARA (10 CFR 20.1101(b)).

10.2 DOSE ASSESSMENT

This section will discuss criteria for design dose rates in the facility and give an estimate of the annual person-Sievert (person-rem) doses associated with major functions.

Information Required at the Time of Construction Authorization–Provide basis, models, and assumptions for calculating the design dose rate values. Include the estimated annual worker occupancy, including number of personnel and duration of occupancy, for radiation areas during normal operations and anticipated operational occurrences. For areas with expected airborne radioactivity concentrations during normal operations and anticipated operational occurrences, provide estimated person-hours of occupancy and estimated inhalation exposures to personnel. Discuss the estimated worker dose as it relates to demonstration of compliance with the ALARA program criteria and to the dose limits and exposure criteria (proposed 10 CFR 63.21(c)(14), proposed 10 CFR 63.111(a)(1), and proposed 10 CFR 63.111(b)(1) [64 FR 8640]). Provide reference to Chapter 7 for details on projected doses within the restricted area as a result of normal operations, plus Category 1 DBE and Category 2 DBE releases. Provide reference to Chapters 4, 5, 6, and 9 for discussion of regulatory design base sources.

Identify worker occupational and public dose limits and the dose-limiting provisions of 10 CFR 20 that apply to the MGR. Identify the additional dose criteria of proposed 10 CFR 63.111(a)(2) and proposed 10 CFR 63.111(b)(1) (64 FR 8640), and Interim Guidance Section 111(b)(2) (Dyer and Horton 1999). Discuss the estimated annual dose to the public due to direct radiation exposure at the boundary of the restricted area (proposed 10 CFR 63.111(b)(2) [64 FR 8640]).

Identify the location and expected annual dose to the individual (real) member of the public as a result of releases during normal operation in relation to the ALARA constraint of 0.1 mSv (10 mrem) per year total effective dose equivalent (TEDE) for air emissions from normal operation (10 CFR 20.1101(d)). Provide a summary of estimated doses to the worker and the public. Provide reference to Chapter 7 for details of calculation and compliance assessment for the doses to the public at the preclosure controlled area boundary. Provide reference to Chapters 4, 5, 6, and 9 for discussion of regulatory design bases sources.

Information Required at the Time of Update to the License Application to Receive and Possess–Provide examples of changes made during planning or design review for reducing projected doses.

10.3 OPERATIONAL RADIATION PROTECTION PROGRAMS

This section will address the DOE commitment to implementing a radiation protection program for the MGR that meets the applicable requirements of 10 CFR 20.1101 through 10 CFR 20.2206 as required by the preclosure performance objective of proposed 10 CFR 63.111(a)(1) (64 FR 8640).

Information Required at the Time of Construction Authorization–State that the operational radiation protection programs for the MGR will demonstrate compliance with proposed 10 CFR 63.21(c)(14) and proposed 10 CFR 63.71(a) (64 FR 8640). State that the DOE will implement health physics programs that will comply with the requirements in 10 CFR 20.1101 through 10 CFR 20.2206. Identify the planned programs. Identify the administrative organization of the programs, including the authority and responsibility of these positions. State that a program will be implemented for calculating offsite doses, including criteria for radiological effluent monitoring, meteorological monitoring, and environmental surveillance

(10 CFR 20.1302; 10 CFR 20.1301; Interim Guidance Section 111(a)(2) [Dyer and Horton 1999]). Identify training criteria that address qualification requirements for radiation protection staff. Provide reference to Section 11.3 for discussion on the MGR training program. Identify the process that will implement the requirement for managing radiation protection records (10 CFR 20, Subpart L) and reports (10 CFR 20, Subpart M, and proposed 10 CFR 63.71(a) [64 FR 8640]). Provide reference to Section 11.4 for description of the MGR records management program. State that the DOE will implement a management program for site-generated low-level radioactive waste (LLW) (10 CFR 20, Subpart K). Provide reference to Chapter 9 for description of the LLW process systems (proposed 10 CFR 63.21(c)(14) [64 FR 8640]).

Information Required at the Time of Update to the License Application to Receive and Possess–Discuss the minimum qualifications for staff positions responsible for health physics programs.

10.3.1 Equipment, Instrumentation, and Facilities

This section will discuss radiation protection support facilities or areas, equipment, and instrumentation necessary to support the implementation of radiation protection programs.

Information Required at the Time of Construction Authorization–State that appropriate facilities, instruments, and equipment will be provided for radiation protection program tasks (proposed 10 CFR 63.21(c)(14) [64 FR 8640]). To the extent that information is available, describe adequacy of planned facilities, instrumentation, and equipment.

Information Required at the Time of Update to the License Application to Receive and Possess–Discuss the respiratory protective equipment, protective clothing, and portable and laboratory technical equipment and instrumentation. Identify the instrument storage, calibration, and maintenance facilities. Describe the types of detectors and monitors and methods of calibration for the technical equipment and instrumentation. Identify health physics facilities (e.g., locker rooms, shower rooms, offices, and access control stations), laboratory facilities for radioactivity analyses, protective clothing, respiratory protective equipment, decontamination facilities, and other contamination control equipment and areas that will be available (proposed 10 CFR 63.21(c)(14) [64 FR 8640]).

10.3.2 Procedures

This section will describe the methods and procedures necessary for conducting radiation protection activities, and related records, reports, and training.

Information Required at the Time of Construction Authorization–State that radiation protection procedures will be developed. Identify procedures and methods of operation that will be developed for ensuring that occupational radiation exposures will be ALARA and satisfactory radiation protection program activities can be safely executed (proposed 10 CFR 63.21(c)(14) [64 FR 8640]). Provide reference to Section 11.2 for discussion of the MGR program to manage and control procedure development.

Information Required at the Time of Update to the License Application to Receive and Possess–Describe the policy, methods, and procedures for conducting radiation protection activities. Include a discussion of the procedures used for dosimetry, surveys, monitoring, in-service inspections, spent nuclear fuel and HLW handling, LLW processing, access control, normal operations, performance confirmation activities, routine maintenance, and sampling and calibration that are specifically related to assuring the radiation exposures will be ALARA and satisfy the requirements of 10 CFR 20.1501, 10 CFR 20.1502, 10 CFR 20.1601, 10 CFR 20.1602, 10 CFR 20.1901, 10 CFR 20.1902, 10 CFR 20.1903, 10 CFR 20.1904, 10 CFR 20.1906, and 10 CFR 20.1801. Identify the methods and procedures for calculation of offsite doses, including implementation of radiological effluent monitoring, meteorological monitoring, and environmental surveillance programs (proposed 10 CFR 63.21(c)(14) [64 FR 8640], 10 CFR 20.1301, and 10 CFR 20.1302). In addition, identify the controls that will be established to ensure that corrective action is initiated in response to effluent releases and radiation exposures (proposed 10 CFR 63.21(c)(14) [64 FR 8640]).

10.4 REFERENCES

The following references were used to develop this chapter of the TGD. For the LA, this section will contain the references used to develop this chapter of the LA.

References Cited

Dyer, J.R. and Horton, D.G. 1999. "Interim Guidance Pending Issuance of New Nuclear Regulatory Commission (NRC) Regulations for Yucca Mountain, Nevada." Letter from J.R. Dyer (DOE/YMSCO) to D.R. Wilkins (CRWMS M&O), June 18, 1999, OL&RC:AVG:1435, with enclosure, "Interim Guidance Pending Issuance of New NRC Regulations for Yucca Mountain." ACC: MOL.19990712.0039.

Codes, Standards, and Regulations

10 CFR (Code of Federal Regulations) 20. Energy: Standards for Protection Against Radiation. Readily available.

64 FR (Federal Register) 8640. Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, Nevada. Proposed rule 10 CFR 63. Readily available.

INTENTIONALLY LEFT BLANK

CHAPTER 11. CONDUCT OF OPERATIONS AND RELATED TOPICS

This chapter of the License Application (LA) provides a description of the program for the conduct of operations, and related topics, at the monitored geologic repository (MGR). The authors will produce a chapter that will enable the U.S. Nuclear Regulatory Commission (NRC) to determine, with reasonable assurance, that repository organization and staffing, procedures, training, records maintenance and retention, emergency planning, and plans for retrieval will result in safe operation consistent with NRC requirements. Authors preparing the LA shall use this document for guidance and must read the Introduction and Appendix B before writing their respective sections.

Proposed 10 CFR 63.24(a) (64 FR 8640) recognizes that all information may not be available at the time of docketing the LA. The guidance in this chapter differentiates the information required at the time of construction authorization (CA) from that required at the time of license to receive and possess high-level radioactive waste and spent nuclear fuel. In the following sections, information identified as "Information Required at the Time of Construction Authorization" will be provided at the time the LA is submitted for docketing. In the future, information that was provided at the time of CA for docketing will be updated, and information developed since the time of docketing the LA will be added as noted in this guidance. The additional information that will be provided in the future is identified as "Information Required at the Time of Update to the License Application to Receive and Possess."

Compliance Program Guidance Packages have been prepared related to the contents of this chapter. Ensure all applicable requirements, as identified in the appropriate compliance program guidance packages, are addressed.

11. PURPOSE AND SUMMARY

Provide an introductory discussion of the purpose of Chapter 11 of the LA. Include a summary of the information contained in the chapter and a discussion of its organization.

11.1 ORGANIZATIONAL STRUCTURE AND MANAGEMENT

State the purpose of this section, which is to demonstrate that the U.S. Department of Energy (DOE) has in place adequate organization, management, and staff to comply with the requirements of proposed 10 CFR 63 (64 FR 8640) and other regulations referenced within it, and to demonstrate the ability of the DOE to construct and operate the repository through all phases in a manner that protects the environment, safety, and health of the public and workers. This section should identify and describe the structure, functions, and responsibilities of the construction and operating organization. This organizational description satisfies the requirements of proposed 10 CFR 63.21(c)(22)(i), proposed 10 CFR 63.21(c)(22)(ii), and proposed 10 CFR 63.21(c)(22)(iii) (64 FR 8640).

Refer to the *Safety Analysis Report for the INEL TMI-2 Independent Spent Fuel Storage Installation* (INEL 1996) and the *Centralized Interim Storage Facility Topical Safety Analysis Report* (DOE 1998) for examples of an organizational description involving both the DOE and a Management and Operating Contractor.

11.1.1 U.S. Department of Energy

11.1.1.1 Organization

Information Required at the Time of Construction Authorization–Describe the organizational structure of DOE offices having jurisdiction over the planning and construction aspects of the project and include a functional organizational chart. Specify the minimum experience and qualification requirements for the key positions discussed. Show lines of authority, oversight, responsibility, and the ability of the DOE to exercise management and administrative control of activities at the repository. Include both offsite and onsite DOE organizations. Describe the envisioned functional organization that will support operations at the facility. State that the organizational structure for the preoperational and operational aspects will be described by the time of the update to support the LA to receive and possess.

Information Required at the Time of Update to the License Application to Receive and Possess–Describe the organizational structure of DOE offices having jurisdiction over the preoperational and operational aspects of the project, including physical security. Provide an organizational chart showing the structure and position titles. Show lines of authority, oversight, responsibility, and the ability of the DOE to exercise management and administrative control of activities at the repository. Include both offsite and onsite DOE organizations.

11.1.1.2 Interrelationships with Contractors

Information Required at the Time of Construction Authorization–State that the DOE will construct and operate the facility with the aid of a qualified contractor. Describe the working interrelationships and organizational interfaces among DOE offsite and onsite organizations and the major contractors responsible for construction.

Information Required at the Time of Update to the License Authorization to Receive and Possess–Describe the working interrelationships and organizational interfaces among DOE offsite and onsite organizations and the major contractors responsible for management and operational control. Discuss the contractual agreements and other controls that exist among the organizations to ensure safe operation of the repository (NRC 1989, Section 9.1.1.3).

11.1.1.3 Liaison with Outside Organizations

Discuss arrangements with outside organizations, including those providing technical expertise concerning installation design and construction, process and equipment selection or development, and safety evaluations.

11.1.2 Contractors

11.1.2.1 Contractor In-House Organization

Information Required at the Time of Construction Authorization–Describe the organizational structure of the major contractors responsible for the design and construction of the repository facility. Include the major contractors working on design and preparation or that are projected to have roles in the construction or operation of the repository. Provide an

organizational chart and show lines of authority, oversight, and responsibility. If the proposed organization is different for various phases of the repository, describe each applicable organization to the extent it is known. State that the contractor organizational structure for operation of the facility will be provided by the update of the LA to support the LA to receive and possess.

Information Required at the Time of Update to the License Application to Receive and Possess–Describe the organizational structure of the major contractors with responsibility for operation of the facility. Include the major contractors that are working on preoperational testing and operation. Provide an organizational chart and show lines of authority, oversight, and responsibility.

The information below applies to both the CA and the LA to receive and possess.

Personnel Functions, Responsibilities, and Authority–List the key onsite management personnel and functions, including authority and responsibilities. Provide a description of delegations of authority, including specific succession to responsibility for overall repository operation in the event of absences, incapacitation, or other emergencies.

Personnel Qualification Requirements–Describe the qualification requirements for designated key positions in terms of specific functions or positions. Provide a description of mandatory training, ongoing training, and experience levels for designated key positions. If required qualifications of key positions/functions have not already been obtained, provide a schedule that ensures completion of required qualifications.

11.1.2.2 Technical Staff

Information Required at the Time of Construction Authorization–Describe the contractor technical staff supporting facility siting, engineering, and construction. Provide a description of the technical staff duties, responsibilities, and authority. Include minimum requirements for staff qualifications, educational background, and technical experience. Describe any arrangements made for technical support to be provided by outside consultants (NRC 1989, Section 9.1.1.4). Establish that the experience and qualification of technical staff personnel meet the requirements of ANSI/ANS-3.1-1993 (ANSI/ANS 1993). State that the contractor technical staff training program to support operations of the facility will be provided by the time of the LA to receive and possess.

Information Required at the Time of Update to the License Application to Receive and Possess–Describe the contractor technical staff supporting facility operations. Provide a description of the technical staff duties, responsibilities, and authority. Include minimum requirements for staff qualifications, educational background, and technical experience. Describe any arrangements made for technical support to be provided by outside consultants (NRC 1989, Section 9.1.1.4). Establish that the experience and qualification of technical staff personnel meet the requirements of ANSI/ANS-3.1-1993 (ANSI/ANS 1993).

11.1.2.3 Contractor Operating Organization

Information Required at the Time of Construction Authorization–Describe the facility functional groups that will operate the facility and illustrate with a functional organizational chart. State for all activities important to safety that: responsibility is well defined; there will be clear lines of authority to the facility manager; the functional areas of radiation protection, security, quality assurance (QA), and training will be separate from the operating organization; and sufficient managerial depth will be available on all shifts to provide qualified backup in all instances.

Information Required at the Time of Update to the License Application to Receive and Possess–Describe the organization responsible for safe operation of the repository. Include key supervisory personnel and those operating positions requiring specific qualifications or certifications. Demonstrate for all activities important to safety that: responsibility is well defined; there are clear lines of authority to the facility manager; the functional areas of radiation protection, security, QA, and training are separate from the operating organization; and sufficient managerial depth is available on all shifts to provide qualified backup in all instances.

The following information applies to both the CA and the LA to receive and possess.

Personnel Qualification Requirements–Describe the minimum qualification requirements (experience, training, and education) for specific functions or positions listed above. Provide a description of mandatory training, ongoing training, and required certifications for facility positions/functions. If required qualifications and certifications of key positions/functions are not already obtained, provide a schedule that ensures completion of required training and certification.

11.2 PROCEDURES

This section will describe the procedural administrative control processes for developing, reviewing, approving, controlling, and changing all classifications of procedures. It will describe the plans for developing the procedures that will be used for the programmatic implementation of QA requirements during design and construction and later for safe conduct of repository operations during the preclosure period, including startup and testing. It identifies those procedures and how they will be used.

Appendix B of 10 CFR 50, Subpart G of proposed 10 CFR 63 (64 FR 8640), and Interim Guidance Section 31(a)(6) (Dyer and Horton 1999) contain requirements applicable to the process for identifying, preparing, and maintaining administrative and implementing procedures. Interim Guidance Section 31(a)(6) (Dyer and Horton 1999) requires the NRC to consider the DOE plan for developing operating procedures to assist in making a determination of reasonable assurance that there is not an unreasonable risk to the health and safety of the public concerning the use of radioactive materials. In addition, 10 CFR 71 and 10 CFR 72, while not directly applicable, may provide useful insight to the author.

11.2.1 General Procedures

Information Required at the Time of Construction Authorization–Explain the various processes that will be put in place to develop the administrative and implementing procedures governing quality-affecting activities during design, construction, and operation of the repository. Show that the new procedures and changes to procedures will be provided with reviews and approvals adequate to ensure proper implementation of regulatory requirements. Reference the regulatory, commitment, and guidance documents used to develop each of the procedural processes.

State that all fuel handling operations will be conducted by procedures that have been prepared, reviewed, approved, and tested. State that the fuel handling tests will be conducted using mockups. State that all maintenance for structures, systems, or components (SSCs) important to safety will be carried out in accordance with procedures that have been prepared, reviewed, approved, and tested.

Provide preliminary lists by subject areas of procedures to be developed for fuel handling operations, maintenance, and administration. State that the plans for procedures covering off-normal events will be discussed in the update of the LA to receive and possess.

Information Required at the Time of Update to the License Application to Receive and Possess–Explain the various processes that have been put in place to develop the administrative and implementing procedures governing quality-affecting activities during design, construction, and operation of the repository. Show that new procedures and changes to procedures will be provided with reviews and approvals adequate to ensure proper implementation of regulatory requirements. Reference the regulatory, commitment, and guidance documents used to develop each of the procedural processes.

11.2.2 Other Procedures

List by subject area other procedures to be developed. As an example, use those listed in NRC Regulatory Guide 1.70 (NRC 1978), Section 13.5.2.2, as appropriate to MGR activities.

11.2.3 Off-Normal Procedures

Information Required at the Time of Construction Authorization–State that off-normal procedures will be developed to address areas such as: fire brigades, equipment alarms, and site emergencies (e.g., flash flood, tornado, and radioactive particulate release). State that a detailed discussion on the development of these procedures will be provided in the update to the LA to receive and possess.

Information Required at the Time of Update to the License Application to Receive and Possess–List the off-normal procedures to be developed and provide schedules for their development and testing. Include the following subject areas: fire brigades, equipment alarms, site emergencies (e.g., flash flood, tornado, and radioactive particulate release). Summarize the contents of each procedure. State that procedures for responding to security threats are to be withheld from public disclosure.

11.3 TRAINING

This section provides a description of the Office of Civilian Radioactive Waste Management MGR training program. This training program will meet the requirements of proposed 10 CFR 63 (64 FR 8640), Subpart H, Training and Certification of Personnel, for issuance of an NRC license. Specifically, this section will meet the requirements of proposed 10 CFR 63.151, General Requirements; proposed 10 CFR 63.152, Training and Certification Program; and proposed 10 CFR 63.153, Physical Requirements (64 FR 8640).

This section develops, documents, and implements a training program to ensure that facility personnel are trained and qualified to safely and effectively perform job requirements. Specifically, this section addresses training for personnel performing operations, maintenance, and supervision and management of those systems and components that have been identified as important to safety in the safety analysis and the LA.

This section describes how the MGR training program integrates each of the following program elements:

- Training organization
- Development and qualification of training staff
- Personnel selection and entry-level requirements
- Facility personnel training
- Qualification and certification process
- Training records.

11.3.1 Training Organization

Information Required at the Time of Construction Authorization–State that a training organization will be established to ensure facility personnel are adequately trained and qualified for their positions. State that the training department will develop training programs using a systematic approach to training to ensure the qualification of personnel in accordance with proposed 10 CFR 63, Subpart H requirements (64 FR 8640). State that a description of the training organization will be provided by the update of the LA to receive and possess. State that line management is responsible to ensure their personnel are trained and qualified to safely and effectively perform job requirements. State that only certified personnel or other personnel under the direct visual supervision of a certified individual shall operate systems and components important to safety. State that supervisory personnel who direct operations important to safety shall be certified in such operations.

Information Required at the Time of Update to the License Application to Receive and Possess–Provide a basic training department organizational chart. Provide a description of the training department personnel functions, responsibilities, and authority.

11.3.2 Development and Qualification of Training Staff

Information Required at the Time of Construction Authorization–State that the training department will ensure that instructors have the qualifications, knowledge, and experience for the

subject matter that they are assigned to teach. State that the instructor qualification training will include a continuing instructional skills training program to maintain, improve, and update the knowledge and teaching skills of the existing training staff. State that the instructor qualification training program will be provided by the update to the LA to receive and possess.

Information Required at the Time of Update to the License Application to Receive and Possess–Describe the instructor qualification-training program that has been put in place.

11.3.3 Personnel Selection and Entry-Level Requirements

Information Required at the Time of Construction Authorization–State that procedures or policies defining personnel selection and entry-level requirements for facility personnel will be developed by the update of the LA to receive and possess. State that entry-level requirements will specify the minimum education, technical, and experience requirements necessary for an employee to meet the job requirements.

State that any condition that might cause impaired judgement or motor coordination will be considered in the selection of personnel for activities that are important to safety (proposed 10 CFR 63.153 [64 FR 8640]).

Information Required at the Time of Update to the License Application to Receive and Possess–Establish that facility management is responsible for hiring or transferring only personnel who meet the entry requirements of a position. Criteria for exemptions to required training for those individuals who may be qualified to perform job requirements will be based on prior experience or training.

11.3.4 Facility Personnel Training

Information Required at the Time of Construction Authorization–This section describes the facility training program. This program is based on a systematic approach to training, as well as the established hazards of the facility. Explain that the five essential elements of a systematic approach to training are:

- Systematic analysis of the jobs to be performed
- Learning objectives derived from the analysis of the job that describe performance after training
- Training design, development, and implementation that is based on the learning objectives
- Evaluation of trainee mastery of the objectives during training
- Evaluation and revision of the training based on the performance of trained personnel in the job setting.

State that the initial and continuing training programs will be implemented to ensure facility personnel are qualified to perform job requirements in a safe and efficient manner.

State that the facility training program will allow program change and update to incorporate changes to the facility, procedures, regulations, and industry/facility operating experience and to the performance of trained facility personnel.

State that the following types of training programs will be developed: general employee, technician and maintenance, health physics, emergency response, technical staff, management and supervisory, and underground safety training.

Information Required at the Time of Update to the License Application to Receive and Possess–Briefly describe the facility training programs listed above. Also include any other training programs specific to the repository that have been identified.

11.3.5 Qualification and Certification Process

Information Required at the Time of Construction Authorization–State that a program for the qualification and certification of facility operators, supervisors, and maintenance personnel performing operations with SSCs important to safety, including requalification training and exceptions to training requirements, will be developed. State that only those certified personnel or other personnel under the direct visual supervision of a certified individual shall operate system and components important to safety (proposed 10 CFR 63.151 and proposed 10 CFR 63.152 [64 FR 8640]). State that the qualification and certification training program will comprise the following components used to validate personnel knowledge and operating capabilities after completion of the appropriate training program:

- Written examinations
- Oral examinations
- Operating examinations–on-the-job training.

Information Required at the Time of Update to the License Application to Receive and Possess–Briefly describe the programs that have been identified above.

11.3.6 Training Records

Information Required at the Time of Construction Authorization–State that an administrative program for the maintenance of training, qualification, and certification records for MGR organization personnel will be developed. State that training records will be maintained to allow the review of course content, schedule, trainee attendance, evaluation results, and employee qualification. State that the training records will be maintained consistent with the Records Management System as discussed in Section 11.4.

Information Required at the Time of Update to the License Application to Receive and Possess–Briefly describe the program for maintaining training records.

11.4 RECORDS

Proposed 10 CFR 63.21(c)(17) (64 FR 8640) requires that a description be provided for the program used to maintain the records described in proposed 10 CFR 63.71 and proposed

10 CFR 63.72 (64 FR 8640). This section describes the program established to manage and preserve records of geologic repository construction and operations.

11.4.1 Records Management Program

Describe the program, as required in proposed 10 CFR 63.21(c)(17), for maintaining records of geologic repository activities, including: a complete history of receipt, handling, storage, and disposition of radioactive wastes (Interim Guidance Section 71(b) [Dyer and Horton 1999]); construction records as discussed in 10 CFR 63.72(b) (64 FR 8640); record requirements of Interim Guidance Section 44(c)(1) (Dyer and Horton 1999); and records connected with the licensed activity that may be required by the conditions of the license or by rules, regulations, and orders of the NRC, as authorized by the Atomic Energy Act and the Energy Reorganization Act (proposed 10 CFR 63.71(a) [64 FR 8640]). State that procedures implementing MGR records requirements will be developed in accordance with the Office of Civilian Radioactive Waste Management QA program.

State that records will be retained in accordance with an approved project procedure that specifies the appropriate retention schedule. Describe the measures taken to ensure that the record is legible throughout the specified retention period. Describe the safeguards in place to prevent tampering and loss of records.

11.4.2 Record Preservation

Information Required at the Time of Construction Authorization–State that a system or process will be established for the permanent preservation of site records to meet the requirements of proposed 10 CFR 63.51(a)(3)(ii), proposed 10 CFR 63.71(b), and proposed 10 CFR 63.72(a) (64 FR 8640).

Information Required at the Time of Update to the License Application to Receive and Possess–Describe the system or process that is planned for the permanent preservation of site records, including site data collected prior to site characterization, data from site characterization experiments, in situ tests, construction records, and performance confirmation records. Show how the system meets the requirements of proposed 10 CFR 63.51(a)(3)(ii) and proposed 10 CFR 63.71(b) (64 FR 8640) for ensuring that records are available for future generations.

11.5 ADMINISTRATIVE PROGRAMS

This section describes the administrative programs that will address changes, tests, and experiments, as well as the employee concerns program.

11.5.1 Changes, Tests, and Experiments

Information Required at the Time of Construction Authorization–State that a systematic program for implementing Interim Guidance Section 44(b) (Dyer and Horton 1999) will be developed.

Information Required at the Time of Update to the License Application to Receive and Possess–Discuss the program to review, evaluate, and document changes, tests, and experiments

as described in Interim Guidance Section 44(b) (Dyer and Horton 1999) to the geologic repository operations area.

11.5.2 Employee Concerns Program

Describe the employee concerns program to be in place at the time the LA is written and prior to initiation of construction and operations. State that the program complies with proposed 10 CFR 63.9 (64 FR 8640). Specifically describe how the program will ensure protected activities listed in proposed 10 CFR 63.9(a)(1) (64 FR 8640) will not be subject to discrimination. State that the program will be documented by appropriate procedures and that employees will receive formal training on the program prior to the start of construction. State that an NRC Form 3 will be posted as required by proposed 10 CFR 63.9(e)(1) (64 FR 8640).

11.6 REPOSITORY CONSTRUCTION INTERFACE

Describe the repository construction interface controls, including the interfaces between the construction and performance confirmation activities and the construction and waste emplacement activities when both are proceeding simultaneously. Details should be specific enough to enable the NRC to understand and agree to the validity of the concepts, but not so specific as to require LA updates to address minor changes in the concepts. However, this section should emphasize an integrated discussion of the construction process rather than focus on specific repository configuration items.

11.7 STARTUP ACTIVITIES AND TESTING

As required by proposed 10 CFR 63.21(c)(22)(iv) (64 FR 8640), describe the plans for startup activities and startup testing. Provide an outline discussing the hierarchy of planned supporting test documents and plans. State that the detailed test documents and plans will reside outside the LA and will address construction tests, pre-operational tests, and startup tests. Provide a general description of the test program organization. Provide a high-level schedule for completing the testing activities.

11.8 SURVEILLANCE AND PERIODIC TESTING

This section provides a description of the surveillance and periodic testing plan as required by proposed 10 CFR 63.21(c)(22)(v) (64 FR 8640).

Information Required at the Time of Construction Authorization–Provide a general description of the type of surveillance and periodic testing that will be performed as required by proposed 10 CFR 63.21(c)(22)(v) (64 FR 8640). State that a program to perform surveillance and periodic testing for equipment important to safety and waste isolation will be developed by the update to the LA to receive and possess. Include a description of the purpose of the program and provide examples of the types of surveillance and periodic tests to be performed. State that a program will be developed to identify the types of tests to be performed. State that the surveillance requirements for equipment important to safety will be contained in the license specifications, a separate document as discussed in Section 11.10, License Specifications.

Information Required at the Time of Update to the License Application to Receive and Possess–Describe the surveillance and test program that has been developed.

11.9 MAINTENANCE

This section describes the facility maintenance program in accordance with proposed 10 CFR 63.21(c)(22)(v) (64 FR 8640), which requires that information be provided on the "Plans for conduct of normal activities, including maintenance...."

Information Required at the Time of Construction Authorization–State that a maintenance program will be developed to inspect, test, and maintain SSCs important to safety to ensure their continued functioning and readiness. The maintenance program, when developed, will address the organizational structure, procedures, work control processes, and program monitoring.

Information Required at the Time of Update to the License Application to Receive and Possess–Describe the aspects of the maintenance program discussed below.

Maintenance Organization and Management–Describe the planned maintenance organization. Include key positions and associated job responsibilities. Show how the maintenance organization interfaces with the facility operating and technical organization.

Maintenance Personnel Qualification and Training–Provide a reference to Section 11.3 for training program details.

Maintenance Procedures–Describe the operational procedures to maintain SSCs important to safety and retrievability at the geologic repository operations area, both surface and underground. Describe how procedures will be used in the performance of maintenance activities. Identify and briefly describe the classes of procedures, schedules, and plans for maintenance of SSCs important to safety, retrievability, and isolation for all surface facilities, shafts, and ramps, and the underground facility. Provide a reference to Section 11.2 for details associated with the facility procedure program.

Types of Maintenance–Briefly discuss the types of maintenance planned at the facility. Include preventative, corrective, predictive, and maintenance surveillance. Show how each maintenance type supports reliable operation of SSCs important to safety and waste isolation.

Work Control Process–Briefly describe the planned maintenance program work control process. Include the following items:

- Planning
- Scheduling
- Post-maintenance testing
- Maintenance history update.

Maintenance Recordkeeping–Describe the system for recording, maintaining, and utilizing equipment maintenance records. Show how the system contributes to performance of effective maintenance and facility and equipment reliability.

Maintenance Program Monitoring, Assessment, and Feedback–Describe the methods planned for assessing the safety and effectiveness of the repository maintenance program. Describe how feedback from maintenance and operations personnel will be used to improve maintenance program effectiveness.

State that monitoring and assessment of the repository maintenance program satisfies the intent of the NRC maintenance rule as presented in 10 CFR 50.65 and Regulatory Guide 1.160 (NRC 1997), as applicable, and provide support for the statement. The maintenance rule requires that the holder of an operating license monitor the performance or condition of SSCs against licensee-established goals, in a manner sufficient to provide reasonable assurance that they are capable of fulfilling their intended functions.

11.10 LICENSE SPECIFICATIONS

This section describes the requirements associated with license specifications. License specifications will be proposed by the DOE to the NRC after the LA for CA is submitted. The NRC will then review the license specifications and issue them with the license to receive and possess waste. Introduce this section by stating that the purpose of license specifications is to provide the variables, conditions, or other items that govern the licensed operation of the facility. The license specifications will define operating limits and controls, monitoring instruments and control settings, surveillance requirements, design features, and administrative controls that ensure the facility is operated in a manner that protects the health and safety of the public and the facility workers.

11.10.1 Subjects of License Specifications

As required by proposed 10 CFR 63.21(c)(13) (64 FR 8640), variables, conditions, or other items determined to be probable subjects of license specifications are to be identified in the LA for CA. State that special attention has been given to those items that may significantly influence the final design. Justify the selection of these items by providing the bases for inclusion. Include applicable design criteria or the results of repository safety analyses.

The probable subjects of license specifications should be identified to the fullest extent possible, including numerical values and other pertinent data. Also, identify the applicable sections of the LA that develop the details and bases of the license specifications. In summary, the license specifications must be derived from facility safety analysis and include all aspects of repository operations that are important to safety, retrievability, and waste isolation. It is likely that at the time of the LA to receive and possess, this information will be removed, as a draft of the proposed license specifications should be prepared, as discussed in Section 11.10.2.

11.10.2 License Specification Development Process

State that the purpose of this section is to describe the license specification development process.

Information Required at the Time of Construction Authorization–Proposed 10 CFR 63.42(a) and proposed 10 CFR 63.43 (64 FR 8640) specify that a license issued in accordance to those parts shall contain license specifications and conditions. Therefore, the LA

should include a discussion of the license specification development process. The DOE will provide the proposed license specifications to the NRC in a separate document from the LA after the LA at CA and before the LA to receive and possess. The NRC will review license specifications proposed by the DOE, revise and augment them as deemed appropriate, and issue the final license specifications as an NRC document.

The LA should include an example license specification. The suggested content of an individual license specification is as follows:

- Title: Write the name or description of the limit or control.
- Limiting Condition for Operation: Identify the lowest functional capability or performance level of equipment required for safe operation.
- Applicability: Define systems or operations to which the control or limit applies.
- Action: Define actions to be taken if the control or limit is exceeded.
- Surveillance requirements: Describe the maintenance and tests to be performed with specified frequency.
- Bases: Describe the bases for each control or limit, and explain they must contain a summary of the information in sufficient depth to indicate the completeness and validity of the supporting information and to provide justification for the control or limit.

Proposed 10 CFR 63.43(b)(6) (64 FR 8640) specifies requirements for administrative controls that will also be included in the license specifications. Administrative controls are defined as "the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure that the activities at the facility are conducted in a safe manner and in conformity with the other license specifications." Discuss how the administrative controls will be developed from the related material in the LA.

Information Required at the Time of Update to the License Application to Receive and Possess–Provide current status of license specification development process.

11.11 EMERGENCY PLANNING

This section provides an overview of the facility emergency plan. The emergency plan must meet the requirements of Subpart I, Emergency Planning Criteria, for issuance of an NRC license. Specifically, this section will meet the requirements of proposed 10 CFR 63.21(c)(16) (64 FR 8640), which requires the LA to provide a description of the plan for responding to, and recovering from, radiological emergencies that may occur at any time before permanent closure and decontamination or dismantlement of surface facilities. Interim Guidance Section 161 (Dyer and Horton 1999) states that the emergency plan shall be based on the criteria of 10 CFR 72.32(b). State that the emergency plan will be submitted to the NRC in a separate document from the LA between the time of LA at CA and LA to receive and possess, and is summarized in this section. State that the intent of this section is to demonstrate that emergency planning by the DOE will adequately protect public health and safety during repository

operation. The emergency planning information should provide evidence of preliminary planning for emergency preparedness directed at situations involving actual or potential radiological hazards.

Information Required at the Time of Construction Authorization–State that the outline and content of the emergency plan follows the outline in 10 CFR 72.32(b). Present a high-level summary of the plan. List the types of radiological accidents covered in the plan. Discuss the need, if any, for offsite assistance to respond to the kinds of radiological accidents listed. Commit to training offsite response organizational personnel and to having periodic drills. Commit to coordinate preparation of the emergency plan with offsite responders, and consider their comments in developing the final emergency plan.

Emphasize in detail those practices or features of emergency planning for the geologic repository that are substantially different from elsewhere in the nuclear industry. Provide a sufficient level of detail to give the NRC reviewers confidence that the project will implement an emergency planning program that will meet or exceed requirements.

Information Required at the Time of Update to the License Application to Receive and Possess–Discuss comments received from offsite response organizations on the initial emergency plan and the summary information on how they were incorporated. Reference letters of agreement between the DOE and offsite response organizations. Discuss training completed for offsite personnel. State that a schedule (five-year) for conducting quarterly communications checks with offsite response organizations and two-year onsite exercises to test responses to simulated emergencies has been prepared. Address offsite organization participation in the two-year exercises. Commit to conduct post-exercise evaluations. Address how future changes to the plan will be handled with respect to response organization and the public.

11.12 WASTE RETRIEVAL, REMOVAL, AND ALTERNATE STORAGE

Describe the plan for retrieval, removal, and alternative storage of radioactive wastes if the geologic repository is proven unsuitable for their disposal. This description is required in accordance with proposed 10 CFR 63.21(c)(19) (64 FR 8640).

Information Required at the Time of Construction Authorization–Describe the concept of waste retrieval, including reasons for waste movement and retrieval. Discuss conceptual ideas for full and partial retrieval under various normal and abnormal scenarios involving the natural and engineered barriers. The abnormal scenarios should include leaking waste packages. Describe the ventilation and transportation capabilities required to support retrieval. Describe the expected retrieval environment for both normal and abnormal conditions. Provide a description of the alternate storage location. Figures should be included to illustrate the alternate storage layout, including the conceptual storage unit and ideas for equipment, buildings, and processes used to retrieve the waste. The need for any additional shielding should also be discussed.

State that no action will be taken that would make emplaced high-level radioactive waste irretrievable or would substantially increase the difficulty of retrieving waste unless the NRC authorizes a license amendment to permit such action as discussed by proposed

10 CFR 63.46(a)(1) (64 FR 8640). Provide a reference to Chapter 3 for the evaluation of the likelihood of the various natural events, such as earthquakes and water ingress, that could cause the need for, or impede the ability to accomplish, waste retrieval. If backfill is selected as an option, include a discussion on the impact of backfill on the retrieval plan.

Proposed 10 CFR 63.111(e)(3) (64 FR 8640) discusses a reasonable schedule for retrieval. State the assumed schedule for waste retrieval and include the timeframes associated with various scenarios for retrieval.

Information Required at the Time of Update to the License Application to Receive and Possess–Provide any new information on the various scenarios considered and the means to cope with them. Incorporate any results of research programs conducted regarding waste retrieval.

11.13 SAFETY QUESTION RESOLUTION

This section describes the program for resolving safety questions. As required by proposed 10 CFR 63.21(c)(21) (64 FR 8640), identify, describe, and discuss those safety features or components for which further technical information is required to confirm the adequacy of design. The LA must be as complete as possible using information that is available at the time of submission. Provide an explanation, with reference to information not supplied at the time of submitting the CA, of why such information was not available.

Information Required at the Time of Construction Application–Include the following information in this section:

- For Quality Level 1, 2, and 3 SSCs, identify and describe the research and development programs that will be conducted to resolve any safety questions.
- Describe the specific technical information that must be obtained to demonstrate acceptable resolution of the program.
- Describe the program in sufficient detail to show how the information will be obtained, or cross-reference those sections of the LA in which the information is provided.
- Provide a schedule of completion of the program as related to the projected startup date of repository operation.
- Discuss the design alternatives or operational restrictions available in the event that the results of the program do not demonstrate acceptable resolution of the problem.

Include a description of any special technical information development programs undertaken to establish the final design and/or to demonstrate the conservatism of the design. Include a discussion of any programs that will be conducted during operation to demonstrate the acceptability of contemplated future changes in design or operation.

Provide a schedule indicating when items are expected to be resolved. Make a commitment to include resolved questions in amendments to the LA.

Information Required at the Time of Update to the License Application to Receive and Possess–For any safety questions still unresolved, describe the plans to resolve them. Delete the information related to the program designed to resolve safety questions after it is shown that the scheduled research and development requirements have been satisfied and properly reflected in the design chapters.

11.14 ACTIVITIES UNRELATED TO DISPOSAL OF RADIOACTIVE WASTE

Proposed 10 CFR 63.21(c)(22)(vii) (64 FR 8640) requires that any activities conducted at the facility for purposes other than disposal of radioactive waste be discussed. Support activities conducted at the facility for the disposal of site-generated waste is not included in this discussion.

Information Required at the Time of Construction Authorization–State that any activities not related to disposal of radioactive waste are not anticipated to occur at the facility site. State that if it is determined that such activities may occur, they will be reflected in a future update to the LA, which will include their effects on waste isolation and repository performance.

Information Required at the Time of Update to the License Application to Receive and Possess–If there are no activities that have been identified that meet the criteria of proposed 10 CFR 63.21(c)(22)(vii) (64 FR 8640), provide a statement to this effect. If there are activities that meet these criteria, include them in the update and their effects on waste isolation and repository performance.

11.15 DECONTAMINATION OR DISMANTLEMENT OF SURFACE FACILITIES

Proposed 10 CFR 63.21(c)(22)(vi) (64 FR 8640) requires that plans for permanent closure and plans for the decontamination or dismantlement of surface facilities be provided in the LA. State that detailed plans will be developed to address decontamination or dismantlement of the surface facility prior to the license application for closure of the facility. State that the requirements of 10 CFR 20, Subpart E regarding decontamination and dismantlement of facilities will be followed. Provide a reference to Chapter 4 for the description of design features intended to facilitate permanent closure or dismantlement of surface facilities. Discuss any conceptual plans for decontamination or dismantlement and the timeframe for accomplishing them to the extent they are known.

11.16 REFERENCES

The following references were used to develop this chapter of the technical guidance document. For the LA, this section will contain the references used to develop this chapter of the LA.

References Cited

ANSI/ANS (American National Standards Institute/American Nuclear Society) 1993. *Selection, Qualification, and Training of Personnel for Nuclear Power Plants.* ANSI/ANS-3.1-1993. La Grange Park, Illinois: American Nuclear Society. TIC: 235767.

DOE (U.S. Department of Energy) 1998. *Centralized Interim Storage Facility Topical Safety Analysis Report*. BA0000000-01717-5700-00017 REV 01. Two volumes. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.19990212.0117.

Dyer, J.R. and Horton, D.G. 1999. "Interim Guidance Pending Issuance of New Nuclear Regulatory Commission (NRC) Regulations for Yucca Mountain, Nevada." Letter from J.R. Dyer (DOE/YMSCO) to D.R. Wilkins (CRWMS M&O), June 18, 1999, OL&RC:AVG:1435, with enclosure, "Interim Guidance Pending Issuance of New NRC Regulations for Yucca Mountain." ACC: MOL.19990712.0039.

INEL (Idaho National Engineering Laboratory) 1996. *Safety Analysis Report for the INEL TMI-2 Independent Spent Fuel Storage Installation*. Docket No. 72-20, Rev. 0. Idaho Falls, Idaho: U.S. Department of Energy, Idaho Operations Office. TIC: 233637.

NRC (U.S. Nuclear Regulatory Commission) 1978. *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants LWR Edition*. Regulatory Guide 1.70, Rev. 3. Washington, D.C.: U.S. Nuclear Regulatory Commission. TIC: 222636.

NRC 1989. *Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage)*. Regulatory Guide 3.48, Rev. 1. Washington, D.C.: U.S. Nuclear Regulatory Commission. Readily available.

NRC 1997. *Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*. Regulatory Guide 1.160, Rev. 2. Washington, D.C.: U.S. Nuclear Regulatory Commission. Readily available.

Codes, Standards, and Regulations

10 CFR (Code of Federal Regulations) 20. Energy: Standards for Protection Against Radiation. Readily available.

10 CFR 50. Energy: Domestic Licensing of Production and Utilization Facilities. Readily available.

10 CFR 71. Energy: Packaging and Transportation of Radioactive Material. TIC: 238422.

10 CFR 72. Energy: Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste. TIC: 242777.

64 FR (Federal Register) 8640. Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, Nevada. Proposed rule 10 CFR 63. Readily available.

INTENTIONALLY LEFT BLANK

CHAPTER 12. PERFORMANCE CONFIRMATION PROGRAM

This chapter of the License Application (LA) provides a description of the performance confirmation program. The authors will produce a chapter that will demonstrate to the U.S. Nuclear Regulatory Commission (NRC) that the performance confirmation program meets the requirements of Subpart F of proposed 10 CFR 63.21(c)(20) (64 FR 8640). Authors preparing the LA shall use this document for guidance and must read the Introduction and Appendix B before writing their respective sections.

12. PURPOSE AND SUMMARY

This section of Chapter 12 should provide a brief introduction to the chapter and informs the reader why we are presenting the information provided. A brief summary of Chapter 12 should also be provided in this section. State that the performance confirmation program meets the requirements of Subpart F of proposed 10 CFR 63.1(c)(20) (64 FR 8640). State that the performance confirmation program provides data important to parameters and conceptual models used in the performance assessment that indicate, where practicable, whether actual subsurface conditions encountered and changes in those conditions during construction and waste emplacement operations are within the limits assumed in the performance assessment (Interim Guidance Section 131(a)(1) [Dyer and Horton 1999]).

State that the performance confirmation program is limited to key observable parameters important to postclosure analyses (Interim Guidance Section 131(a)(2) [Dyer and Horton 1999]). Provide a reference to Chapter 8, where important data from the total system performance assessment (TSPA) for the LA is listed.

Present summary information supporting the conclusions that the performance confirmation program provides the means for the program to inform the NRC of changes in the field conditions being monitored and the subsequent need for changes in design or construction methods in an appropriate and timely fashion, in accordance with proposed 10 CFR 63.132(d) (64 FR 8640).

Provide the conclusions reached regarding the adequacy of the performance confirmation program. State that data collected during performance confirmation will be used to update models and evaluate total system performance as required. Also, state that the results of monitoring and analysis will be compared to the predicted system response (proposed 10 CFR 63.132(d) [64 FR 8640]).

Provide a general description of the monitored geologic repository (MGR) phases.

12.1 PERFORMANCE CONFIRMATION APPROACH

This section provides the performance confirmation program approach. In general terms, describe the performance confirmation program. Discuss the performance confirmation program as it applies to the stages of repository operation, from site characterization to permanent closure, to demonstrate that the program complies with proposed 10 CFR 63.131(b) (64 FR 8640).

Describe the process for selecting key performance confirmation parameters. State that, as a criterion for selecting these parameters, the parameter must be measured, monitored, observed, or tested to confirm that actual subsurface conditions fall within the limits assumed in the licensing review, and is in accordance with proposed 10 CFR 63.131(a)(1) (64 FR 8640). This section shall also contain a statement that the performance confirmation approach will include a process of identifying parameter limits and a process of identifying when recorded values exceed the defined limits (Interim Guidance Section 131(a)(1) [Dyer and Horton 1999]). State that the selection is based on TSPA sensitivity analysis, process model validation requirements and data needs, and specific regulatory requirements (Interim Guidance Section 132(c) [Dyer and Horton 1999]). Explain the selection criteria for parameters associated with borehole and access seals, backfill, and the thermal-interaction effects of the waste packages (WPs), backfill, rock, and groundwater (Interim Guidance Section 133(a) [Dyer and Horton 1999]).

Demonstrate that in situ and other onsite tests do not adversely affect the ability of the natural and engineered elements of the geologic repository to meet the performance objectives for the repository (proposed 10 CFR 63.131(d)(1) [64 FR 8640]).

12.2 PERFORMANCE CONFIRMATION FOR THE NATURAL BARRIERS

This section describes the performance confirmation program for the natural barriers. Describe the performance confirmation program specifically related to the natural barriers.

Include details of any in situ monitoring, geologic mapping, laboratory and field testing, and in situ experiments planned to confirm design assumptions and parameters and to monitor and evaluate changes in the baseline condition of parameters that could affect the performance of the geologic repository (proposed 10 CFR 63.131(c) [64 FR 8640]). Describe how the performance confirmation program confirms that the natural barriers, assumed to function as barriers, are functioning as anticipated (Interim Guidance Section 131(a)(2) [Dyer and Horton 1999]). Discuss the specific parts of the *Performance Confirmation Plan* (CRWMS M&O 1997) that deal with the natural barrier system and how the plan demonstrates compliance with Subpart F of proposed 10 CFR 63 (64 FR 8640). Describe the performance confirmation program components specifically related to the subsurface conditions that are being monitored. State that the subsurface conditions being monitored will be evaluated against the design assumptions used in the TSPA (proposed 10 CFR 63.132(b) [64 FR 8640]).

12.3 REPOSITORY PERFORMANCE CONFIRMATION

This section describes the performance confirmation program for repository testing. Describe the performance confirmation program specifically related to repository testing. Include details of any in situ monitoring, geologic mapping, laboratory and field testing, and in situ experiments planned to confirm design assumptions and parameters, and to monitor and evaluate changes in the baseline condition of parameters that could affect the performance of the geologic repository (proposed 10 CFR 63.131(c) [64 FR 8640]). Describe the performance confirmation program components specifically related to the subsurface conditions that are being monitored. State that the performance confirmation activities in the geologic repository operations area will be performed during the early or developmental stages of construction, and that a program for in

situ testing of features like borehole seals, shaft seals, and backfill will be conducted (Interim Guidance Section 133(a) [Dyer and Horton 1999]).

Provide a description of all performance confirmation testing that will be conducted for the shafts and ramps. Describe the performance confirmation program for the shafts and ramps, from site characterization to immediately prior to permanent closure. This period will include full-scale testing to evaluate the effectiveness of seals, grouts, plugs, and backfill (Interim Guidance Section 133(c) [Dyer and Horton 1999]). Include a discussion of the test sections established to test the effectiveness of shaft and ramp seals (proposed 10 CFR 63.133(d) [64 FR 8640]).

Make a commitment that in situ testing will be implemented as soon as possible during repository construction. Provide a reference to Section 12 for a description of the MGR phases (proposed 10 CFR 63.133(b) [64 FR 8640]).

Describe the performance confirmation program for the underground facility, from site characterization to immediately prior to permanent closure. This period will include full-scale testing to evaluate the effectiveness of seals, grouts, plugs, and backfill (Interim Guidance Section 133(a) [Dyer and Horton 1999]).

Define the baseline data used for borehole seal locations and the testing methods planned for borehole seals (proposed 10 CFR 63.133(d) [64 FR 8640]).

Make a commitment that in situ testing will be implemented as soon as possible during repository construction, and indicate that in situ monitoring of the thermomechanical response extends to the start of closure (proposed 10 CFR 63.132(e) and proposed 10 CFR 63.133(b) [64 FR 8640]). Provide a reference to Section 12 for a description of the MGR phases and to the design chapters for specific descriptions of the design considerations required to implement this aspect of the performance confirmation program (proposed 10 CFR 63.111(d) [64 FR 8640]).

12.4 PERFORMANCE CONFIRMATION FOR THE WASTE PACKAGE AND ENGINEERED BARRIER SYSTEM

This section describes the performance confirmation program for the WP and the engineered barrier system (EBS). Describe the performance confirmation program components specifically related to the subsurface conditions that are being monitored. Specifically address how the geotechnical and design parameters are monitored and evaluated against design assumptions (proposed 10 CFR 63.132(b) [64 FR 8640]). Provide a reference to the design chapters for specific descriptions of the design considerations required to implement this aspect of the performance confirmation program (proposed 10 CFR 63.111(d) [64 FR 8640]). Provide a reference to the performance objective for the EBS doses in proposed 10 CFR 63.113(b) (64 FR 8640) as necessary.

12.4.1 Waste Package Performance

Describe performance confirmation activities for the WP, including the definition of baseline data. State that this work will ensure that the engineered structures, systems, and components

required for repository operation, and designed or assumed to operate as barriers after permanent closure, function as intended and anticipated (Interim Guidance Section 131(a)(2) [Dyer and Horton 1999]).

12.4.1.1 Waste Form

Describe the waste form to which the performance confirmation program applies, and state why it is representative of the wastes to be emplaced (proposed 10 CFR 63.134(a) [64 FR 8640]).

12.4.1.2 In Situ Waste Package Monitoring

Describe the in situ monitoring program. Also describe how the WPs chosen for the in situ program are representative of all the emplaced WPs (proposed 10 CFR 63.134(a) [64 FR 8640]).

12.4.1.3 External Waste Package Environment

Describe the environment of the in situ WPs chosen. Describe the environmental conditions of the WPs chosen for the in situ monitoring program, and indicate how these conditions are representative of the emplacement drifts in general (proposed 10 CFR 63.134(b) [64 FR 8640]).

12.4.1.4 Laboratory Testing

Provide a summary of the WP monitoring program intended to satisfy proposed 10 CFR 63.134(c) (64 FR 8640). State how the laboratory experiments focus on the internal condition of the WPs and how the experiments duplicate the environment experienced by the emplaced WPs (proposed 10 CFR 63.134(c) [64 FR 8640]).

12.4.1.5 Duration of Post-Emplacement Waste Package Monitoring

State that the WP monitoring extends to the start of closure and that monitoring will continue up to the time of approval of the license amendment for permanent closure and the start of closure (proposed 10 CFR 63.134(d) [64 FR 8640]).

12.4.2 Engineered Barrier System (Excluding the Waste Package) Performance

Describe performance confirmation activities for the EBS (excluding the WP), including defining baseline data for the emplacement horizon and in situ monitoring of conditions in the emplacement drifts. Briefly list components, and reference Chapter 6 as appropriate. Indicate that the repository discussion in that section excludes the EBS described in Section 12.4. State that this work will ensure that the engineered structures, systems, and components required for repository operation, and designed or assumed to operate as barriers after permanent closure, function as intended and anticipated (Interim Guidance Section 131(a)(2) [Dyer and Horton 1999]). Include details of any in situ monitoring, geologic mapping, laboratory and field testing, and in situ experiments planned to confirm design assumptions and parameters and to monitor and evaluate changes in the baseline condition of measurable parameters that could affect the performance of the geologic repository (proposed 10 CFR 63.131(c) [64 FR 8640]).

If backfill is included in the LA design, state that performance confirmation activities for the underground facility include defining baseline data for backfill, in situ testing of backfill (i.e., the backfill test section), and field tests to evaluate backfill placement technology (i.e., constructability tests). State that in situ backfill tests will check the thermal-interaction effects of the WPs, backfill, rock, and groundwater as required by proposed 10 CFR 63.133(a) (64 FR 8640). State that field tests will compare the effectiveness of backfill placement and compaction procedures against design requirements before permanent backfill placement is begun (Interim Guidance Section 133(c) [Dyer and Horton 1999]).

12.5 PLANS TO REDUCE AREAS OF UNCERTAINTY

This section describes the performance confirmation program related to the NRC key technical issues (KTIs). Describe specific plans for further testing as part of the performance confirmation program to reduce significant areas of uncertainty related to the KTIs of the NRC. KTIs are defined in a recent NRC annual report (CNWRA 1997).

If the testing program for coupled thermal-hydrologic-chemical (THC) processes on seepage and flow is not complete at the time of the LA, or if sensitivity and uncertainty analyses indicate additional data are needed, describe the identified specific plans to acquire the necessary information as part of the performance confirmation program [ENFE 1.1.5] (NRC 1998a).

Describe how the performance confirmation program assesses whether the natural system and engineered materials are functioning as anticipated with regard to coupled THC effects on seepage and flow [ENFE 1.2.8] (NRC 1998a).

If the testing program for coupled THC processes on the WP chemical environment is not complete at the time of the LA, or if sensitivity and uncertainty analyses indicate additional data are needed, describe the identified specific plans to acquire the necessary information as part of the performance confirmation program [ENFE 2.1.7] (NRC 1998a).

Describe how the performance confirmation program assesses whether the natural system and engineered materials are functioning as anticipated with regard to coupled THC effects on the WP chemical environment [ENFE 2.2.12] (NRC 1998a).

If the testing program for coupled THC processes on the chemical environment for radionuclide release from the EBS is not complete at the time of the LA, or if sensitivity and uncertainty analyses indicate additional data are needed, describe the identified specific plans to acquire the necessary information as part of the performance confirmation program [ENFE 3.1.7] (NRC 1998a).

Describe how the performance confirmation program assesses whether the natural system and engineered materials are functioning as anticipated with regard to coupled THC effects on the chemical environment for radionuclide release from the EBS [ENFE 3.2.12] (NRC 1998a).

If the testing program for coupled THC processes radionuclide transport is not complete at the time of the LA, or if sensitivity and uncertainty analyses indicate additional data are needed,

describe the identified specific plans to acquire the necessary information as part of the performance confirmation program [ENFE 4.1.7] (NRC 1998a).

Describe how the performance confirmation program assesses whether the natural system and engineered materials are functioning as anticipated with regard to coupled THC effects on radionuclide transport in the near field [ENFE 4.2.12] (NRC 1998a).

If the thermal-hydrologic testing program is not complete at the time of the LA submittal, describe the identified specific plans for completion of the testing program as part of the performance confirmation program [TEF 3] (NRC 1998b).

Related to the effects of corrosion processes on the lifetime of the waste packages, describe specific plans for further testing to reduce any significant areas of uncertainty as part of the performance confirmation program [CLST 1.6] (NRC 1998c).

Describe the program of corrosion monitoring and testing of the engineered subsystem components during the performance confirmation period, and how this assures that they are functioning as intended [CLST 1.7] (NRC 1998c).

Related to the effects of materials stability and mechanical failure on the lifetime of the waste package, describe specific plans for further testing to reduce any significant areas of uncertainty as part of the performance confirmation program [CLST 2.6] (NRC 1998c). Describe the program of monitoring and mechanical testing of the engineered subsystem components, during the performance confirmation period, to ensure they are functioning as intended and anticipated in the presence of thermal and stress perturbations [CLST 2.7] (NRC 1998c).

Related to the rate of degradation of spent fuel and the rate at which radionuclides in spent fuel are released from the engineered barrier subsystem, describe specific plans for further testing to reduce any significant areas of uncertainty as part of the performance confirmation program [CLST 3.6] (NRC 1998c).

Related to the rate of degradation of high-level waste glass and the rate at which radionuclides in high-level waste glass are released from the engineered barrier subsystem, describe specific plans for further testing to reduce any significant areas of uncertainty as part of the performance confirmation program [CLST 4.6] (NRC 1998c).

Describe any plans for monitoring radionuclide release from the WP during the performance confirmation period to assure that assumptions and calculations regarding high-level radioactive waste glass dissolution and radionuclide release are appropriately substantiated [CLST 4.7] (NRC 1998c).

Related to the effects of alternate engineered barrier subsystem design features on the waste package lifetime and radionuclide release from the EBS, describe specific plans for further testing to reduce any significant areas of uncertainty as part of the performance confirmation program [CLST 6.6] (NRC 1998c).

For aspects of analytical or numerical models related to thermal-mechanical analyses of the underground facility for which long-term experimental data are needed, refer the reader to documents containing detailed plans and procedures for verification and validation during performance confirmation [RDTME 3.1.7] (NRC 1998d).

12.6 ANALYSIS OF CHANGES FROM PERFORMANCE CONFIRMATION BASELINE

This section describes how the performance confirmation program will detect and deal with changes from the baseline conditions that have been established by the performance assessment in accordance with proposed 10 CFR 63.131(d)(3) (64 FR 8640) Describe the performance confirmation baseline information for parameters that could significantly affect the postclosure performance of a geologic repository (proposed 10 CFR 63.131(d)(2) [64 FR 8640]). These parameters will come from the performance assessment. Describe the plan for monitoring and analyzing changes from the baseline condition of measurable parameters that could significantly affect the postclosure performance of a geologic repository. The format in the *Performance Confirmation Plan* (CRWMS M&O 1997), Section 3.4, provides a suitable format for this section. Include a section on NRC communications and reporting (Interim Guidance Section 132(a) [Dyer and Horton 1999]). State that changes in design or construction methods are made based on the performance confirmation program. Describe how the channels of communication between the U.S. Department of Energy (DOE) and the NRC during performance confirmation have been established. Describe any agreements between the NRC and the DOE that specifically cover activities during performance confirmation. Describe the method of the performance confirmation program to issue a series of reports documenting progress during construction and monitoring. Show that the program has been planned and implemented to satisfy the requirements of proposed 10 CFR 63.132(d) (64 FR 8640).

12.7 REFERENCES

The following references were used to develop this chapter of the technical guidance document. For the LA, this section will contain the references used to develop this chapter of the LA.

References Cited

CNWRA (Center for Nuclear Waste Regulatory Analyses) 1997. *NRC High-Level Radioactive Waste Program Annual Progress Report Fiscal Year 1996*. NUREG/CR-6513, No. 1. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.19970813.0082.

CRWMS (Civilian Radioactive Waste Management System) M&O (Management and Operating Contractor) 1997. *Performance Confirmation Plan.* B00000000-00841-4600-00002 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19980204.1022.

Dyer, J.R. and Horton, D.G. 1999. "Interim Guidance Pending Issuance of New Nuclear Regulatory Commission (NRC) Regulations for Yucca Mountain, Nevada." Letter from J.R. Dyer (DOE/YMSCO) to D.R. Wilkins (CRWMS M&O), June 18, 1999, OL&RC:AVG:1435, with enclosure, "Interim Guidance Pending Issuance of New NRC Regulations for Yucca Mountain." ACC: MOL.19990712.0039.

NRC (U.S. Nuclear Regulatory Commission) 1998a. *Issue Resolution Status Report Key Technical Issue: Evolution of the Near-Field Environment*. Rev. 1. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.19981106.0144.

NRC 1998b. *Issue Resolution Status Report Key Technical Issue: Thermal Effects on Flow*. Rev. 1. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.19990317.0357.

NRC 1998c. *Issue Resolution Status Report Key Technical Issue: Container Life and Source Term*. Rev. 1. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.19990105.0081.

NRC 1998d. *Issue Resolution Status Report Key Technical Issue: Repository Design and Thermal-Mechanical Effects*. Rev. 1. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.19981130.0219.

Codes, Standards, and Regulations

64 FR (Federal Register) 8640. Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, Nevada. Proposed rule 10 CFR 63. Readily available.

CHAPTER 13. LAND OWNERSHIP AND CONTROL

This chapter contains general guidance for preparing Chapter 13 of the License Application (LA), which describes the U.S. Department of Energy (DOE) program for managing ownership and control of interests in land. The LA must provide sufficient information for the U.S. Nuclear Regulatory Commission to make a reasonable assurance determination that the DOE has identified the interests in property it has obtained or will obtain to meet the requirements for land ownership and control (proposed 10 CFR 63.21(c)(15); proposed 10 CFR 63.121(c)(1) and (2), and proposed 10 CFR 63.121(a)(2)(i)-(iii) [64 FR 8640]; Interim Guidance Section 121(a)(1), and Interim Guidance Section 121(b) [Dyer and Horton 1999]). In other words, the required level of detail is that needed to demonstrate compliance with regulations. Demonstrating compliance involves providing sufficient technical basis to allow the NRC to determine that there is reasonable assurance that the repository can and will be designed, constructed, and operated without unreasonable risk to the health and safety of the public, and to demonstrate that there is reasonable assurance that postclosure performance objectives will be met, consistent with the regulations. The guidance presented in this chapter applies for docketing of the LA at the time of construction authorization and at the time of updating the LA to receive and possess high-level radioactive waste. All the information provided in this chapter must be presented in the LA at the time of docketing. The information will be confirmed to be correct or updated as appropriate when the LA update is submitted.

Authors preparing the LA shall use this document for guidance and must read the Introduction and Appendix B before writing their respective sections.

13. PURPOSE AND SUMMARY

This section will present a summary of the DOE land ownership and control strategy to comply with the requirements of proposed 10 CFR 63.21(c)(15), proposed 10 CFR 63.121(a)(2)(i)-(iii), and proposed 10 CFR 63.121(c)(1) and (2) (64 FR 8640); Interim Guidance Section 121(a)(1) and Interim Guidance Section 121(b) (Dyer and Horton 1999). State the purpose of the chapter and discuss the organization of the chapter.

13.1 ACQUISITION OF THE SITE

This section will address how the DOE has acquired or will acquire ownership of the land and other interests in property to comply with Interim Guidance Section 121(a)(1) (Dyer and Horton 1999) and proposed 10 CFR 63.121(a)(2)(i)-(iii) (64 FR 8640). This section will also address how the DOE plans to comply with proposed 10 CFR 63.121(c)(1) (64 FR 8640), as it relates to sufficiency of water rights.

13.1.1 Identification of the Postclosure Controlled Area

This section will define and describe the postclosure controlled area (Interim Guidance Section 121(a)(1) [Dyer and Horton 1999]). Provide complete, detailed, and legal description of the postclosure controlled area that conforms to other accepted methods of land description. Provide complete U.S. Bureau of Land Management master title plats for all sections contained within the postclosure controlled area to ensure they identify existing or proposed title control

and existing encumbrances. Identify the limits of the geologic repository operations area (GROA) and its relationship to the limits of the postclosure controlled area on master title plats, maps, and diagrams (proposed 10 CFR 63.21(c)(15) [64 FR 8640] and Interim Guidance Section 121(a)(1) [Dyer and Horton 1999]). If controls for the GROA are different from those for the preclosure controlled area, describe the extent of the difference (proposed 10 CFR 63.21(c)(15) [64 FR 8640] and Interim Guidance Section 121(a)(1) [Dyer and Horton 1999]).

Identify areas adjacent to the postclosure controlled area. Describe human actions outside the postclosure controlled area considered to have the potential to affect the geologic repository and explain how this potential affects designation of areas of concern[1] (Interim Guidance Section 121(b) [Dyer and Horton 1999]).

13.1.2 Legal Interests

This section will identify and describe the legal interests to be obtained in real property by the DOE. Describe how existing legal interests in the postclosure controlled area will be dissolved. Describe how the DOE has acquired or will permanently withdraw the land that would comprise the GROA and postclosure controlled area (Interim Guidance Section 121(a)(1) [Dyer and Horton 1999]). To demonstrate compliance with the requirements of proposed 10 CFR 63.121(a)(2)(i)-(iii) (64 FR 8640), discuss existing and future surface and subsurface legal interests of the following types:

- Rights arising under general mining laws
- Easements for rights-of-way
- Other rights arising under lease rights of entry, deed, patent, mortgage appropriation, and prescription.

Provide reference to the legal documentation of ownership, including sufficient indices of ownership and control, to satisfy a purchaser of record. Describe any existing or future encumbrances or other surface or subsurface interests of record identified (proposed 10 CFR 63.121(a)(2)(i)-(iii) [64 FR 8640]).

13.1.3 Water Rights

This section will discuss water rights determined to be necessary to operate the monitored geologic repository (MGR) through license termination and to prevent adverse impact on waste isolation. Describe needed permits. Provide schedules and show they are adequate to meet Yucca Mountain Project schedules for GROA construction, development, and waste emplacement. Identify water rights for both the periods of operation and after permanent closure (proposed 10 CFR 63.121(c) [64 FR 8640]).

---

[1] If the site is sufficiently large to make regulation of the surrounding areas unnecessary, refer to Section 13.2 for discussion of the rationale to satisfy the requirements of Interim Guidance Section 121(b) (Dyer and Horton 1999) as it affects the designation of areas of concern.

13.2 ACCESS CONTROLS

This section will describe the precautions that the DOE will take to restrict access to surface areas of the MGR before permanent closure (proposed 10 CFR 63.21(c)(15) [64 FR 8640]). State that the DOE will implement a strategy to regulate land use and water rights outside the postclosure controlled area (Interim Guidance Section 121(b) [Dyer and Horton 1999] and proposed 10 CFR 63.121(c)(2) [64 FR 8640]). If the postclosure controlled area is sufficiently large to make regulation of the surrounding areas unnecessary, provide rationale to satisfy the requirements of Interim Guidance Section 121(b) (Dyer and Horton 1999). Refer to Section 1.6 of the LA for information on the physical protection measures that will limit access to protected areas of the MGR.

This section also will provide a conceptual design description for the system of site markers, including monuments, used to permanently identify the repository following permanent closure and to prevent future human interactions (proposed 10 CFR 63.21(c)(15) [64 FR 8640]). State that the postclosure controlled area and the GROA will be identified by markers designed, fabricated, and emplaced to be as permanent as practicable.

13.3 REFERENCES

The following references were used to develop this chapter of the technical guidance document. For the LA, this section will contain the references used to develop this chapter of the LA.

References Cited

Dyer, J.R. and Horton, D.G. 1999. "Interim Guidance Pending Issuance of New Nuclear Regulatory Commission (NRC) Regulations for Yucca Mountain, Nevada." Letter from J.R. Dyer (DOE/YMSCO) to D.R. Wilkins (CRWMS M&O), June 18, 1999, OL&RC:AVG:1435, with enclosure, "Interim Guidance Pending Issuance of New NRC Regulations for Yucca Mountain." ACC: MOL.19990712.0039.

Codes, Standards, and Regulations

64 FR (Federal Register) 8640. Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, Nevada. Proposed rule: 10 CFR 63. Readily available.

INTENTIONALLY LEFT BLANK

APPENDIX A

CROSS-REFERENCE BETWEEN PROPOSED 10 CFR 63 (64 FR 8640) REGULATIONS AND TECHNICAL GUIDANCE DOCUMENT

APPENDIX A

CROSS-REFERENCE BETWEEN PROPOSED 10 CFR 63 (64 FR 8640) REGULATIONS AND TECHNICAL GUIDANCE DOCUMENT

This appendix provides cross-references between the proposed 10 CFR 63 (64 FR 8640) regulations and the sections in the *Technical Guidance Document for License Application Preparation* (TGD) that provide guidance for demonstrating compliance with those regulations. The section numbers listed in this appendix refer to both the License Application (LA) sections in which compliance is to be demonstrated and the corresponding TGD sections that provide acceptance criteria and guidance for the LA sections.

Not all regulations in proposed 10 CFR 63 (64 FR 8640) actually contain requirements for the licensee. Therefore, notes are provided to identify the reasons why these regulations are not addressed in the TGD. The following notes appear in the cross-reference that follows:

Note 1: Not a requirement for the LA; not addressed in the TGD. No specific U.S. Department of Energy (DOE) action is required in the LA. Regulations that are cross-referenced with this note provide clarifications to proposed 10 CFR 63 (64 FR 8640) and activities required prior to submittal of the LA (e.g., purpose and scope, interpretations, site characterization, and progress reports).

Note 2: Not a requirement for the DOE. The regulations state actions that the U.S. Nuclear Regulatory Commission (NRC) will take to ensure compliance by the DOE (e.g., finding that the construction is complete, inspections, and conditions of construction authorization).

Note 3: Not a requirement for the LA. These regulations are general requirements not applicable to one or more specific sections in the LA (e.g., communications, employee protection, and distribution of the LA).

This appendix also provides cross-references between the DOE Interim Guidance (Dyer and Horton 1999) and the sections in the TGD that provide guidance for demonstrating compliance with this interim guidance.

Proposed 10 CFR 63 (64 FR 8640) Cross-Reference

Regulation	TGD Section(s) Addressing the Regulation
63.1 – 63.16	These regulations are not required to be in the LA since they deal with definitions or clarifications to proposed 10 CFR 63 (64 FR 8640). No specific action is required of the DOE to address these regulations in the LA. These regulations are not addressed in the LA.
63.21(a)	Affected by DOE Interim Guidance, Section 21(a)
63.21(b)(1)	1.1.1, 1.1.3, 1.1.4
63.21(b)(2)	1.2
63.21(b)(3)	Affected by DOE Interim Guidance, Section 21(b)(3)
63.21(b)(4)	1.6
63.21(b)(5)	1.4.1
63.21(c)(1)(i)	3., 3.1.2
63.21(c)(1)(ii)	3.2.1, 3.2.9, 3.3.1, 3.5.1
63.21(c)(1)(iii)	3.4.1, 3.5.4
63.21(c)(1)(iv)	3.1.3, 8.4.1
63.21(c)(2)	2.1, 7., 7.1
63.21(c)(3)	2.3, 2.5, 3.2.9, 4.1.1.1, 4.1.1.2, 4.1.1.3, 5.2.3, 6.1.2.1, 6.1.2.2, 6.1.2.3, 9.1.1.1, 9.1.1.2, 9.1.1.3
63.21(c)(4)(i)	5.2.3, 6.1.2.1, 6.1.2.2, 6.1.2.3
63.21(c)(4)(ii)	2.5, 5.2.3, 6.1.2.1
63.21(c)(5)	3.2.5, 5.2.3, 6.4, 8.4.1
63.21(c)(6)	3.2.9, 8.4.3
63.21(c)(7)	Affected by DOE Interim Guidance, Section 21(c)(7)
63.21(c)(8)	2.1, 8., 8.1.2, 8.5.1
63.21(c)(9)	3.2, 3.3, 3.4, 3.5, 5.2, 5.2.3, 5.5, 5.6, 5.7, 5.8, 6.4, 8.3.1, 8.4.1
63.21(c)(10)	3.2.2, 3.2.12, 3.3.2, 3.4.2, 3.5.2, 5.2.3, 6.4, 8.3.1

Regulation	TGD Section(s) Addressing the Regulation
63.21(c)(11)	Affected by DOE Interim Guidance, Section 21(c)(11)
63.21(c)(12)	1.1.2, 5.1, 7.2.2, 7.4, 7.4.1, 7.4.2, 7.4.4, 7.4.5
63.21(c)(13)	4.1.1.1, 4.1.1.2, 4.1.1.3, 5.2.3, 6.1.2.1, 6.1.2.2, 6.1.2.3, 9.1.1.1, 9.1.1.2, 9.1.1.3, 11.10.1
63.21(c)(14)	4.1.1.1, 4.1.1.2, 4.1.1.3, 6.1.2.1, 6.1.2.2, 6.1.2.3, 9.1.1.1, 9.1.1.2, 9.1.1.3, 10.1, 10.1.1, 10.1.2, 10.2, 10.3, 10.3.1, 10.3.2
63.21(c)(15)	13.1.1, 13.2
63.21(c)(16)	11.11
63.21(c)(17)	Addressed via 63.71 and 63.72
63.21(c)(18)	4.1.1.1, 4.1.1.2, 4.1.1.3, 4.1.1.4, 4.7, 6.3, 6.3.1, 6.3.3, 6.3.4, 9.1.1.1, 9.1.1.2, 9.1.1.3, 9.1.1.4
63.21(c)(19)	5.2.3, 6.2.6, 11.12
63.21(c)(20)	12.
63.21(c)(21)	4.1.1.1, 4.1.1.2, 4.1.1.3, 5.2.3, 5.4.2, 6.1.2.1, 6.1.2.2, 6.1.2.3, 9.1.1.1, 9.1.1.2, 9.1.1.3, 11.13
63.21(c)(22)(i)	1.3, 11.1
63.21(c)(22)(ii)	1.3, 11.1
63.21(c)(22)(iii)	11.1
63.21(c)(22)(iv)	11.7
63.21(c)(22)(v)	11.8, 11.9
63.21(c)(22)(vi)	11.15
63.21(c)(22)(vii)	11.14
63.22	Deals with filing and distribution of the LA. Not addressed in the TGD.
63.23	1.7
63.24	Deals with updating of the LA and EIS. Specific guidance for providing information in the LA at the time of docketing and at the time of LA update to receive and possess HLW is provided in each chapter as appropriate.
63.31(a)(1)	2.

Regulation	TGD Section(s) Addressing the Regulation
63.31(a)(2)	2., 2.1
63.31(a)(3)	2.
63.31(a)(4)	2.
63.31(a)(5)	2.
63.31(a)(6)	Affected by DOE Interim Guidance, Section 31(a)(6)
63.31(b)	1.6
63.31(c)	Deals with actions required of the NRC to issue a CA. No action is required of the DOE. Not addressed in the TGD.
63.32	Deals with conditions that the NRC may impose on the CA. No action is required of the DOE in the LA. Not addressed in the TGD.
63.33	Deals with amendment of the CA. No action is required of the DOE in the LA. Not addressed in the TGD.
63.41(a) – 63.41(d)	Deals with actions required of the NRC to issue a license to receive and possess HLW. No action is required of the DOE in the LA. Not addressed in the TGD.
63.41(e)	1.
63.42(a)	11.10.2
63.42(b) – 63.42 (d)	Deals with mandatory conditions on the license. No action is required of the DOE. Not addressed in the TGD.
63.43	Deals with License Specifications. Addressed at a summary level in Section 11.10.2 of the TGD.
63.44(a)	Deals with explanations of terms relative to changes, tests, and experiments. Not addressed in the TGD.
63.44(b)	Affected by DOE Interim Guidance, Section 44(b)
63. 44(c)(1)	Affected by DOE Interim Guidance, Section 44(c)(1)
63.45	Deals with License Amendment. No action is required of the DOE in LA. Not addressed in the TGD.
63.46(a)	Deals with specific activities requiring amendment to License. No action is required of the DOE in LA except for 63.46(a)(1).
63.46(a)(1)	11.12
63.46(b)	Deals with filing of amendment to license. No action is required of the DOE in LA. Not addressed in the TGD.

Regulation	TGD Section(s) Addressing the Regulation
63.51(a)	Except for 63.51(a)(3)(ii), rest of the regulations in 63.51 deal with amendment to license for permanent closure. No action is required of the DOE in LA.
63.51(a)(3)(ii)	11.4.2
63.52	Deals with termination of license following permanent closure. No action is required of the DOE in LA. Not addressed in the TGD.
63.61 – 63.65	Deal with participation by State Government and affected Indian Tribes in the licensing process. No action is required of the DOE in LA. Not addressed in the TGD.
63.71(a)	1.6, 10.3, 11.4.1
63.71(b)	Affected by DOE Interim Guidance, Section 71(b)
63.72(a)	Affected by DOE Interim Guidance, Section 72(a)
63.72(b)	11.4.1
63.73	Deals with reporting deficiencies found during the design and construction of the repository. No action is required of the DOE in LA. Not addressed in the TGD.
63.74	Deals with tests that the NRC may want the DOE to perform, on an as-needed basis. No action is required of the DOE in LA. Not addressed in the TGD.
63.75	Deals with inspections that the NRC may conduct at YMP. No action is required of the DOE in LA. Not addressed in the TGD.
63.78	1.6
63.101(a)(1)	Deals with action required of the NRC to issue a license. No action is required of the DOE in LA. Not addressed in the TGD.
63.101(a)(2)	Explains the basis for uncertainties that exist in meeting postclosure performance standards. The information contained in this regulation is used in presenting postclosure performance material in the LA.
63.101(b)	Deals with action required of the NRC to issue a CA. No action is required of the DOE in LA. Not addressed in the TGD.
63.102	Deals with a functional overview of technical criteria applicable to the repository. No specific action is required, except for 63.102(h) and 63.102(j), since by meeting other applicable regulations the intent of this regulation is satisfied. Not addressed in the TGD.
63.111(a)(1)	4.1.1.1, 4.1.1.2, 4.1.1.3, 6.1.2.1, 6.1.2.2, 6.1.2.3, 7.2, 7.4.3, 7.8.1, 9.1.1.1, 9.1.1.2, 9.1.1.3, 10.1.1, 10.1.2, 10.2, 10.3
63.111(a)(2)	Affected by DOE Interim Guidance, Section 111(a)(2)
63.111(b)(1)	2.1, 4.1.1.1, 4.1.1.2, 4.1.1.3, 5.2.3, 5.3.3, 5.4.1, 5.5, 5.6, 5.7, 5.8, 6.1.2.1, 6.1.2.2, 6.1.2.3, 7.5.2, 9.1.1.1, 9.1.1.2, 9.1.1.3, 10.1.1, 10.1.2, 10.2

Regulation	TGD Section(s) Addressing the Regulation
63.111(b)(2)	Affected by DOE Interim Guidance, Section 111(b)(2)
63.111(c)(1)	2.1
63.111(c)(2)	2.1
63.111(d)	5.2.3, 5.5, 5.6, 5.7, 5.8, 6.3, 12.3, 12.4
63.111(e)(1)	5.2.3, 5.5, 5.6, 5.7, 5.8, 6.2.6
63.111(e)(2)	6.2.6
63.111(e)(3)	11.12
63.112(a)	2.1, 4.1.1.1, 4.1.1.2, 4.1.1.3, 4.1.1.4, 5.2.3, 6.1.2.1, 6.1.2.2, 6.1.2.3, 9.1.1.1, 9.1.1.2, 9.1.1.3, 9.1.1.4
63.112(b)	Affected by DOE Interim Guidance, Section 112(b)
63.112(c)	2.1, 7.2.1, 7.7.1
63.112(d)	2.1, 7.2.1, 7.5.1
63.112(e)	Affected by DOE Interim Guidance, Section 112(e)
63.112(f)	Affected by DOE Interim Guidance, this regulation has been deleted.
63.113(a)	2.1, 2.2, 3., 3.2, 3.3, 3.5, 5.2, 5.2.2, 5.4.1, 5.5, 5.6, 5.7, 5.8, 8.1.3.3
63.113(b)	2.1, 5.2.2, 5.2.3, 5.4.1, 5.5, 5.6, 5.7, 5.8, 6.1.2.1, 6.1.2.2, 6.1.2.3, 6.4, 8., 8.4.3, 8.6.1, 8.8.1, 12.4
63.113(c)	2.1, 8., 8.1.2, 8.4.1
63.113(d)	Affected by DOE Interim Guidance, Section 113(d)
63.114(a)	8.3.1
63.114(b)	8.3.1
63.114(c)	8.3.1
63.114(d)	8.1.2, 8.4.2
63.114(e)	8.2.2

Regulation	TGD Section(s) Addressing the Regulation
63.114(f)	8.2.2
63.114(g)	8.3.1
63.114(h)	3., 3.2, 3.3, 3.5, 5., 6.4, 8.6.2
63.114(i)	3.2, 3.3, 3.5, 5.2.3, 8.6.3
63.114(j)	3.2, 3.3, 3.5, 5.2.3, 8.6.3
63.115(a)(1)	8.3.1.8
63.115(a)(2)	8.3.1.8
63.115(a)(3)	Affected by DOE Interim Guidance, Section 114(k)
63.115(a)(4)	Affected by DOE Interim Guidance, Section 114(l)
63.115(b)(1)	Affected by DOE Interim Guidance, Section 115(b)(1)
63.115(b)(2)	3.1, 8.4.1
63.115(b)(3)	8.4.1
63.115(b)(4)	8.4.1
63.115(b)(5)	8.4.1
63.121(a)(1)	Affected by DOE Interim Guidance, Section 121(a)(1)
63.121(a)(2)(i)	13.1, 13.1.2
63.121(a)(2)(ii)	13.1, 13.1.2
63.121(a)(2)(iii)	13.1, 13.1.2
63.121(b)	Affected by DOE Interim Guidance, Section 121(b)
63.121(c)(1)	13.1, 13.1.3
63.121(c)(2)	13.1.3, 13.2
63.131(a)(1)	Affected by DOE Interim Guidance, Section 131(a)(1)
63.131(a)(2)	Affected by DOE Interim Guidance, Section 131(a)(2)

Regulation	TGD Section(s) Addressing the Regulation
63.131(b)	12.1
63.131(c)	12.2, 12.3, 12.4.2
63.131(d)(1)	12.1
63.131(d)(2)	12.6
63.131(d)(3)	12.6
63.132(a)	Affected by DOE Interim Guidance, Section 132(a)
63.132(b)	12.2, 12.3, 12.4
63.132(c)	Affected by DOE Interim Guidance, Section 132(c)
63.132(d)	12., 12.1, 12.6
63.132(e)	12.3
63.133(a)	Affected by DOE Interim Guidance, Section 133(a)
63.133(b)	12.3
63.133(c)	Affected by DOE Interim Guidance, Section 133(c)
63.133(d)	12.3
63.134(a)	12.4.1.1, 12.4.1.2
63.134(b)	12.4.1.3
63.134(c)	12.4.1.4
63.134(d)	12.4.1.5
63.141	1.5
63.142	1.5
63.143	1.5
63.151	11.3.5
63.152	11.3

Regulation	TGD Section(s) Addressing the Regulation
63.153	11.3.3
63.161	Affected by DOE Interim Guidance, Section 161
63.171	Deals with actions the NRC may take for violations of the provisions of proposed 10 CFR 63 (64 FR 8640). No action is required of the DOE. Not addressed in the TGD.
63.172	Deals with penalties associated with violations of the provisions of proposed 10 CFR 63 (64 FR 8640). No action is required of the DOE. Not addressed in the TGD.

Source: 64 FR 8640

Interim Guidance Cross-Reference

Interim Guidance Section	TGD Section(s) Addressing the IG Section
Sec. 21(a)	1. (Chapters 1 – 13 implement requirement; LA contains statement only regarding requirement)
Sec. 21(b)(3)	1.6
Sec. 21(c)(7)	2.1, 5.2.3, 8., 8.1.1
Sec. 21(c)(11)	1.5
Sec. 31(a)(6)	2., 11.2
Sec. 44(b)	11.5.1
Sec. 44(c)(1)	11.4.1
Sec. 71(b)	1.6, 11.4.1, 11.4.2
Sec. 72(a)	11.4.2
Sec. 111(a)(2)	4.1.1.1, 4.1.1.2, 4.1.1.3, 6.1.2.1, 6.1.2.2, 6.1.2.3, 7.2, 7.2.3, 7.6.1, 7.6.2, 7.7, 7.8.1, 9.1.1.1, 9.1.1.2, 9.1.1.3, 10.1.1, 10.1.2, 10.2, 10.3
Sec. 111(b)(2)	2.1, 4.1.1.1, 4.1.1.2, 4.1.1.3, 5.2.3, 5.4.1, 5.5, 5.6, 5.7, 5.8, 6.1.2.1, 6.1.2.2, 6.1.2.3, 7.3.1.1, 7.3.1.2, 7.5.3, 7.8.2, 9.1.1.1, 9.1.1.2, 9.1.1.3, 10.2
Sec. 112(b)	2.1, 7.1, 7.2.1
Sec. 112(e)	2.1, 2.4, 4.1.1.1, 4.1.1.2, 4.1.1.3, 5.2.3, 6.1.2.1, 6.1.2.2, 6.1.2.3, 7.3, 7.3.2, 9.1.1.1, 9.1.1.2, 9.1.1.3
Sec. 113(d)	2.1, 8., 8.2.4, 8.5.1, 8.5.2, 8.5.3, 8.8.2
Sec. 114(k)	3.4.5, 8.3.1.1

Interim Guidance Section	TGD Section(s) Addressing the IG Section
Sec. 114(l)	3.2.3, 8.4.1
Sec. 115(b)(1)	8.4.1
Sec. 121(a)(1)	13.1, 13.1.1, 13.1.2
Sec. 121(b)	13.1.1, 13.2
Sec. 131(a)(1)	12., 12.1
Sec. 131(a)(2)	12., 12.2, 12.4.1, 12.4.2
Sec. 132(a)	12.2, 12.6
Sec. 132(c)	12.1
Sec. 133(a)	12.1, 12.3, 12.4.2
Sec. 133(c)	12.3, 12.4.2
Sec. 161	11.11

Source: Dyer and Horton 1999

References Cited

Dyer, J.R. and Horton, D.G. 1999. "Interim Guidance Pending Issuance of New Nuclear Regulatory Commission (NRC) Regulations for Yucca Mountain, Nevada." Letter from J.R. Dyer (DOE/YMSCO) to D.R. Wilkins (CRWMS M&O), June 18, 1999, OL&RC:AVG:1435, with enclosure, "Interim Guidance Pending Issuance of New NRC Regulations for Yucca Mountain." ACC: MOL.19990712.0039.

Codes, Standards, and Regulations

64 FR (Federal Register) 8640. Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, Nevada. Proposed rule 10 CFR 63. Readily available.

APPENDIX B

LICENSE APPLICATION WRITER'S GUIDE

APPENDIX B

LICENSE APPLICATION WRITER'S GUIDE

B.1 INTRODUCTION

The purpose of this appendix is to guide License Application (LA) authors in following correct document structure, format, and style, thereby providing a consistent document while strengthening the production process. This writer's guide and the document it references are the only source of guidance for the structure, format, and style that must be used in writing the LA. The basic organization and guidance on content of the LA is provided in the body of the technical guidance document (TGD). Writers must read and conform to these guidelines before writing their respective LA sections.

This writer's guide is based on the assumption that the LA will be developed for hard-copy distribution. Additional instructions provided in Section B.6 address the possibility that the LA may be published electronically.

B.2 WRITE WITH THE AUDIENCE IN MIND

The primary objective of the LA is to present the U.S. Department of Energy safety assessment for a potential high-level radioactive waste repository at Yucca Mountain. While the LA will be submitted to the U.S. Nuclear Regulatory Commission, there will be other readers, including those in various areas of government, as well as the public. Because the LA will contain complex technical subjects and phenomena, the challenge is to present these ideas in terms that the average reader can understand. The guidelines below must be followed in order to achieve effective writing:

- Use active rather than passive voice wherever possible to produce a stronger and more dynamic document.
- Write in the third person rather than the first person.
- Use short declarative sentences, but vary the sentence length when the content justifies it. Break up long sections into shorter sections whenever possible.
- Avoid superlatives and exaggeration; an understated position is more credible.
- Limit the use of jargon and complex technical expressions, if possible, when providing explanations of complex topics.
- Provide references for assertions of fact or common knowledge, or those derived from the LA itself, which are possibly not self-evident to the intended audience.
- Be certain that the facts are current, valid, and complete.

- Review the document using the Microsoft Spelling and Grammar function and then proofread the text for possible word errors.

- While proofing the text, check for clarity of expression by reading the sentences carefully.

B.3 LICENSE APPLICATION DOCUMENT STRUCTURE

The basic organization of the presentation of material in the LA is provided in the body of the TGD. The document structure is hierarchical, starting with a general subject and leading to more specific subjects at lower levels that support the higher-level topic. Present material in a logical progression. Introductory statements at the beginning of each section will provide topical guidance in establishing the document structure.

B.3.1 Chapters

The LA will have a complete table of contents near the front of the document. The table of contents in the LA will show sections created by chapter authors at greater levels of indention and detail than levels provided in this TGD. If the LA chapter contains figures and tables, then these will appear in the text following their textual reference. However, if the chapter is lengthy, the figures and tables will be placed at the end of each chapter. Also, references will be located at the end of each chapter. A list of titles for figures and tables will appear at the front of the document behind the table of contents section.

B.3.2 Sections

A section is a unit of text residing at the second level of indention. Number the sections sequentially within each chapter, using the chapter number followed by a period, and then the sequential section number. Limit section numbers and titles to those provided in the TGD unless a different organization style is approved, in writing, by the Management and Operating Contractor (M&O) LA Development Manager.

B.3.3 Subsections

Within each chapter, use sections and subsections to organize text. Organizing the document in this way provides direction for dividing topics into units that can be examined individually for review. If there is a subsection within a section, then there must be at least two subsections. Refer to subsections as "Sections" (e.g., Section 2.1.1).

Subsections are located at the third or lower levels of indention. Number subsections with three or four digits, separated by periods (e.g., x.x.x, x.x.x.x), depending upon the level of indention, as follows:

2.1.1
2.1.1.1
2.1.1.2
2.1.2

The subsection structure for the LA will be consistent with the LA structure provided in the TGD, although authors may create more subsections.

Subsection numbering is limited to the fourth level of indention (e.g., 2.1.1.1). If necessary, text can further be divided by using zero-level indention, as described in Section B.3.4.

- Using numbered subsections below the fourth level of indention (e.g., x.x.x.x.x) is not allowed. However, unnumbered zero-level indention (as described in Section B.3.4) is permitted when discussing complex topics (e.g., site characteristics).

- Do not create a new subsection if the section only consists of one or two paragraphs. If it is necessary to provide more detail for the topic, use the zero level of indention headings.

- Be sure that the lower level subsection logically expands upon the higher level subsection.

- Use the zero level of indention headings (see Section B.3.4) to relate text to items that are best described in a list.

B.3.4 Zero Level of Indention

A zero level of indention heading does not require a section number. The heading text, which is bolded and left justified, is embedded within the first paragraph of the zero-level subsection and is followed immediately by an unbolded en dash. The first letter of each word in the subsection title (except articles and prepositions) is uppercase. Because zero-level subsections are not numbered, they do not appear in the table of contents. The following example illustrates a zero level of indention heading.

Example:

> **Characteristics of Earthquake Ground Motions at Yucca Mountain–** To date, earthquake ground motions at Yucca Mountain have been estimated using attenuation.

B.3.5 Lists

Use bullets to set items apart in lists. An item in the list can include one or more complete sentences. However, if one or more list items become lengthy paragraphs, the preferred style is a series of subsections rather than a list. The first letter of the first word of each item in the list is capitalized. The last item ends with a closing punctuation.

B.3.6 References

The LA will be a compilation and summary of previously documented work. Therefore, authors must be meticulous about citing the source or sources for material presented in the LA.

Authors will be required to provide complete information for each cited reference, including a photocopy of the front mater of each reference used in developing text. The information to be provided from the front matter for each reference includes:

- Author or authors
- Publication, approval, or effective date
- Complete formal title of the reference
- Associated document number(s)
- Originating organization
- Revision, version, or edition number
- Publisher
- Location of the publisher or originating organization
- Tracking number.

In addition, authors should provide copies of individual pages, figures, and tables that are cited in the LA, provide complete reference information to help researchers efficiently locate documents in the Records Information System or Technical Information Center, as well as help editors, product integrity personnel, and other support personnel verify titles and other reference information. Authors must send their references to the Reference Center as soon as they are available, as well as indicate which chapter(s) the references will be cited in.

Use the author-date style in reference callouts in text (e.g., CRWMS M&O 1999). In the reference list, use the following general format:

> Author, Date, *Title in italics*, Document number. Place of publication: Publisher's name. Accession number.

Refer to the *Style Manual for the Civilian Radioactive Waste Management System Management and Operating Contractor* (CRWMS M&O 1999) for more specific reference list entries.

Whenever possible, cite specific applicable page numbers, figure numbers, table numbers, etc., from the original source. This adherence to specificity is important for three reasons. First, it facilitates technical verification of references to confirm that the reference logically supports the associated material in the LA. Technical verification of references is unduly complicated if the author does not provide specific page numbers to limit the research required in the cited reference. Second, citations that do not specify particular page numbers in the cited reference have the effect of leaving the entire reference vulnerable for potential litigation. The defensibility of the LA will be impaired by referencing irrelevant material. Third, for copyrighted references, the TIC can usually obtain copyright clearance at a lesser cost for selected pages of a document, while copyright clearance for the entire document (or outright procurement of the entire document) would be considerably more expensive.

Citations in the LA to material that is located on the Internet should be avoided. Internet postings remain on the Internet for a short time, and thus can lead to complications associated with obtaining copyright clearance for such material. Contact the M&O LA Development Manager if citation on the Internet material is justified.

Include either a Records Information System accession number or a Technical Information Center catalog number at the end of each LA reference-list entry. References that are determined to be readily available in libraries, on the Internet, or through government agencies are labeled "Readily available" and do not require an accession number.

In general, the LA should not directly cite data contained in the Technical Data Management System but should cite references that, in turn, cite Technical Data Management System data. In the unusual event that a citation to Technical Data Management System data is considered to be necessary, the associated author must inform the M&O LA Development Manager and include the appropriate Technical Data Management System data tracking number (in lieu of a Records Information System accession number or a Technical Information Center catalog number) at the end of the LA reference-list entry.

Authors must cite the latest revision, version, or edition of each cited reference or justify to the M&O LA Development Manager why an historical revision, version, or edition is needed. Note that copies of historical editions of textbooks and reference manuals are often difficult (or impossible) to buy. Any reference that cannot be obtained by Reference Control cannot be cited by the authors.

B.3.7 Cross-Referencing

Cross-referencing among LA chapters is encouraged to reduce the amount of duplicate information given and to minimize the chance of presenting contradictory information. When cross-referencing another LA author's material:

- Inform the other LA lead author of the cross-reference.
- Use the highlighting feature in Microsoft Word to highlight the cross-reference (in yellow).
- Verify, during final preparation of the chapter for review or submittal, that cross-references still connect conceptually to one another (i.e., that the cross-reference directs the reader to the intended material).
- Use the term "section" rather than "subsection" for all cross-referencing (e.g., Section 3.2.2.5).

B.4 LICENSE APPLICATION DOCUMENT TEXT FORMAT

Use Microsoft Word to prepare your text. Accept the default settings for margins, justification, spacing, font, and tab settings. The Critical Document Production group will make any formatting adjustments necessary to conform to the *Style Manual for the Civilian Radioactive Waste Management System Management and Operating Contractor* (CRWMS M&O 1999).

B.4.1 Style

The Document Development and Production Department and the M&O LA Development Manager are responsible for the overall style and appearance of the LA. Direct any questions regarding style or format to the Document Development and Production Department or the M&O LA Development Manager.

B.4.2 Figure and Table Numbering

Number figures and tables according to the heading at the second level of indentation under which the figure or table appears. In other words, the first figure referred to in Chapter 2, Section 2.3, is Figure 2.3-1. Number figures and tables separately. Unless it is more useful to place figures and tables in the text where they are first referred to, place figures and tables at the end of each chapter—first figures, then tables.

B.4.3 Formatting Figures and Maps

Ensure that all text provided in figures is legible. See the *Style Manual for the Civilian Radioactive Waste Management System Management and Operating Contractor* (CRWMS M&O 1999) for further guidance on how to format figures. All maps proposed for inclusion in the LA must be processed through M&O Technical Data Management. Reduce the number of map scales used if possible. All maps should specify latitude and longitude if appropriate. The preferred projection is Universal Transverse Mercator. However, if necessary, obtain permission from M&O Technical Data Management to use a different projection.

B.4.4 Formatting Tables

Ensure that all text provided in tables is legible. See the *Style Manual for the Civilian Radioactive Waste Management System Management and Operating Contractor* (CRWMS M&O 1999) for further guidance on formatting tables. Identify quality data in tables, as explained in Section 8.5 of the Style Manual for the Civilian Radioactive Waste Management System Management and Operating Contractor (CRWMS M&O 1999).

B.4.5 Units of Measurement

Although the units of the International System of Units (SI) are becoming more common in the United States, some readers do not understand them. Therefore, measurements expressed in SI will be expressed in both SI and English units—first by SI and immediately followed by the English equivalent in parentheses (e.g., 1 m (39.37 in.)), with the following exceptions:

- In citing units from references, the convention used in the reference is followed, with conversions to the other type of unit given in parentheses (e.g., 39.37 in. (1 m)).
- For measurements commonly expressed in English units, such as the diameter of pipes, English units are used without conversion to SI.

- Quantities on maps (e.g., elevations) given in English units are not converted to the SI quantities.

Certain quantities may customarily be expressed in mixed units, such as English and SI (e.g., metric tons heavy metal per acre). This practice is undesirable and should be avoided, if possible. Nevertheless, authors may choose to use mixed units if the use is predominant and if the use of other units would not add clarity or assist in understanding the meaning of the quantity.

B.4.6 Numbers

All numbers that appear before units of measurement are written as numerals. Include a space between the last numeral and the units of measurement (e.g., 114 kW) except for degrees of temperature. Units of measurement are abbreviated when preceded by a numeral (e.g., 50 cm) but are spelled out when standing alone (e.g., "the concentration, measured in milligrams per liter..."). If the number preceding a unit is one or less, the unit is written in the singular (e.g., "0.50 m"). In expressing a range or series of measurements, do not repeat the units. Write "40 to 50°C," "five and ten cm," and "40, 60, or 90 cm."

Whole numbers in text are spelled out if they are less than 10 or if they begin a sentence. Numbers equal to or greater than 10 are written as numerals. If any number in a series is greater than 10, the entire series is written as numerals.

Fractions standing alone are spelled out (e.g., "two-thirds of the site"). Fractions used in a series are not spelled out and are best expressed as decimals (e.g., "3.75, 2.75, and 1.75 m" rather than "3 ¾, 2 ¾, and 1 ¾ m"). Always place a zero before the decimal point of a number less than one (e.g., write "0.75" rather than ".75").

Avoid changing units unnecessarily when reporting different amounts of the same quantity (e.g., changing units of a radiation dose from rem to millirem). Do not include more significant digits than can be supported by the underlying reference.

B.4.7 Other Numeric Conventions

In text, spell out units of measurement except for temperatures. Write "812 watts," "600 picocuries per square meter," and "50°C." When temperature is expressed in kelvins, do not use a degree symbol (e.g., 300 K). The degree symbol (°) may also be used for angles, compass directions, longitude, and latitude. In text, spell out "percent." In tables or figures, use the percent symbol (%).

Use standard abbreviations for units of measure. Do not follow abbreviations with a period (except when abbreviating inch), unless the abbreviation appears at the end of a sentence. Do not abbreviate year, mile, or acre. If the abbreviation is derived from the name of a person (e.g., WK.), it is uppercase; otherwise it is lowercase (e.g., g, ft) with the exception of liter (L). The standard prefixes of scientific notation, such as "m," "c," or "k" for "milli," "centi," and "kilo" are lowercase, with the exception of "Giga" and "Mega," which are uppercase.

When the measure is a compound unit designating the multiplication of one unit by another, multiplication is indicated by a hyphen (e.g., g-cm, W-s). Division is indicated by the slash symbol (e.g., J/mole-K, kcal/m-s-K). Measurements involving powers are written in the exponential form (e.g., 10 m^3, 8.34×10^{-8} kg).

For radioisotopes in text, write cesium-137 instead of ^{137}Cs. In tables, write Cs-137. The superscript form can be used in tables only when there is no room for the longer form.

B.5 STYLE

Refer to the *Style Manual for the Civilian Radioactive Waste Management System Management and Operating Contractor* (CRWMS M&O 1999) for style information applicable to the LA that is not specifically discussed in this writer's guide. Pay particular attention to the information in Chapters 9 and 10 of the Style Manual (CRWMS M&O 1999).

In addition to information in the Style Manual (CRWMS M&O 1999), the following guidance applies.

Syntax–Writers must be particularly alert to syntax and choice of verbs to avoid inadvertently undermining the correctness and quality of the completed work. There is a range of certainty suggested by a writer's choice of words. Writers should use a word that fits the intended meaning but should seek to make word choices based on common American usage. Use high-confidence words whenever possible, such as: illustrates, concludes, shows, resolves, states, demonstrates, indicates, establishes, documents, and proves. Low-confidence words, in contrast, include the following: may, maybe, might, could be, seem, appear, suggest, imply, infer, deduce, expect, assume, conceivable, probably, likely, and possibly. "Relatively" and "significant" are two words that confuse most readers and should be used sparingly, if at all. For example, in the sentence, "The impacts are relatively harmless," the reader may wonder, "Relative to what?" After reading, "The U-series dating technique is significantly better than the U-trend technique," the reader may wonder, "Significant according to what standard?" Words should be chosen carefully. In summary, avoid using words that have subjective or multiple meanings.

Commonly Misused Phrases and Words–Misuse of the following terms frequently occurs.

- "All," "never," and "none" are words that should be used with caution, because their use may overstate a fact or conclusion.
- Avoid the use of "maximize," "minimize," "optimize," and similar words whose meanings are subject to excessively wide interpretation.
- "Data," "media," "phenomena," and "criteria" are plural nouns. The corresponding singular nouns are "datum," "medium," "phenomenon," and "criterion."
- The words "offsite" and "onsite," when written as single words, must be used as adjectives, not as adverbs. "The plans call for onsite processing" is acceptable. "Processing is performed onsite" is not acceptable; instead use "at the site" for "onsite."

- The adverbial phrase "under way," when written as two words, means "in progress" or "in motion." The single word "underway" occurs more rarely; it is an adjective meaning "occurring while in motion."

- "Alternative" means "a choice between two or more things." "Alternate" means "succeeding by turns," such as every other day, or to move in position from one side to the other.

- "Due to" is not used in adverbial prepositional phrases by careful writers; it is not a substitute for "because of." Use it only when "due to" clearly modifies a noun. "The machine broke due to improper oiling" is not acceptable; "a failure due to improper oiling" is acceptable.

- The phrase "the maximum individual" appears in regulations about exposure to radiation. Although it cannot always be avoided, its use is objectionable, not only because it is graceless, but also because it does not mean what it seems to mean. Few readers will know that the "individual" is not necessarily a person. Like other technical phrases, this one must be carefully defined if it must be used. Once defined, it can be avoided by the use of a less jarring phrase, like "the maximum individual dose."

- The slash symbol (/) should be used only to denote division in units of measurement. Do not use the slash symbol to mean "and" (e.g., and/or).

Vague Words or Phrases–Words used imprecisely or improperly can confuse readers. Do not use the following vague words or phrases:

- "Anticipate." This is not a synonym for "expect"; it should be used to imply "giving advance thought" to a subject or idea.

- "Based on..." This phrase frequently appears without anything to modify, as in, "Based on the reported data, the committee concluded that no action was necessary." Make sure the phrase modifies something if it must be used.

- "Conservative." Writers often use this word to describe analyses designed intentionally to overestimate risks or adverse impacts. When the word is used to describe an analysis, however, it is necessary to explain which parts of the analysis produce the overestimates. Giving such a complete definition of the word usually removes the need for it.

- "Consider" and "factor." These words are vague, although "factor" does have a precise meaning in mathematics. Writers incorrectly use "factor" to mean "criterion" and "design specification," and "consider" to mean "something to think about."

- "Facility." This word usually conveys little information; define or describe it more clearly in a larger context.

- "Orders of magnitude." This phrase is almost incomprehensible to readers who do not use technical jargon frequently. Write "one ten-thousandth of x" instead of "four orders of magnitude smaller than x."

- Bureaucratic jargon. Careful readers stumble over "officialese," such as "prior to," "implement," "viable," "at this point in time," and a proliferation of "-ize" and "-wise" suffixes. Though some of these words and phrases have precise meanings, they are pretentious. Do not use them.

- The "-ologies" words. The indiscriminate use and creation of words ending in "-ology" leads to imprecise writing. In careful use, the suffix expresses the theory or study of something. "Technology," a fuzzy word that usually means "methods" or "techniques," should be avoided. Do not write "the hydrology of the site"; write "the water flowing through the site" or "the hydraulic system at the site" or another phrase that conveys that meaning. Do not use "methodology" to mean "methods."

B.6 ELECTRONIC DOCUMENT PUBLISHING

In order to meet proposed U.S. Nuclear Regulatory Commission regulation on the Licensing Support Network, the LA will be published electronically (on the Internet and on CD-ROM), as well as in paper format. The Internet version will contain both associative and referential links. The associative link allows the reader to move quickly to other sections within the document or to the document's glossary or list of acronyms. The referential link allows the reader to move electronically to the Yucca Mountain project documents cited in the LA.

B.7 COPYRIGHTED INFORMATION

In general, sources protected by copyright may be quoted in accordance with accepted research practices. That is, under the "fair-use" rule of copyright law, an author may make limited use of another author's work without asking permission. However, because of the legal implications of quoting information from copyrighted sources, such quotations must be evaluated by the Technical Information Center Supervisor to determine whether permission will be required; if it is, they will obtain copyright clearance. Complete reproductions of text, tables, or figures from copyrighted sources cannot be used in the LA until the necessary permissions are obtained by the Technical Information Center.

B.8 GRAPHICS STANDARDS

Details related to graphics standards are specified in Section 7.1, Graphic Preparation, *Style Manual for the Civilian Radioactive Waste Management System Management and Operating Contractor* (CRWMS M&O 1999).

B.9 REFERENCES

References Cited

CRWMS (Civilian Radioactive Waste Management System) M&O (Management and Operating Contractor) 1999. *Style Manual for the Civilian Radioactive Waste Management System Management and Operating Contractor.* B00000000-01717-3500-00004 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990824.0240.

INTENTIONALLY LEFT BLANK

Responses to NRC's Requests for Additional Information on DOE's* Disposal Criticality Analysis Methodology Topical Report, *YMP/TR-004Q, Revision 0

The following are DOE's draft responses to the NRC's requests for additional information (RAIs) on the *Disposal Criticality Analysis Methodology Topical Report.*

Chapter 1.0 Introduction

1-1 Explain the basis for the following statement: "Present information is that HLW will not contain sufficient amounts of fissile material to pose a criticality risk, even in the absence of any criticality control material. Therefore, the only foreseen application of this analysis methodology to HLW will be to demonstrate this fact for a few worst-case configurations of moderator and geometry."

The sources for the "present information" are not provided. The validity of These statements can only be verified after the indicated information and Demonstration analyses have been submitted.

The "present information" that describes the material specifications of the HLW glass waste form is preliminary (Stout and Leider 1991, Table 6-14). When the HLW compositions are finalized, the validity of the statement regarding the insufficiency of fissile material will be verified. If, as expected, the content of the HLW material is insufficient to pose a credible risk of criticality (i.e., there is not a sufficient mass of fissionable material in a waste package), then the analysis methodology will be used to demonstrate that fact for a few worst-case configurations of moderator and geometry. If, on the other hand, the content of the HLW material can pose a credible risk of criticality (i.e., there is a sufficient mass of fissionable material in a waste package), then the proposed methodology will be applied to the HLW just as it is applied to all other waste forms.

The last two sentences in the footnote were for informational purposes only. This statement is not important for justifying the methodology and can be removed. The two sentences will be removed in a revision to the Topical Report.

Stout, R. B. and Leider, H. R. 1991. *Preliminary Waste Form Characterization Report.* Livermore, California: University of California/LLNL. ACC: MOL.19940726.0118.

Section 1.2 Objective

1-2 Explain why the Criticality Consequence criterion refers to consequences of only a single criticality event.

Certain classes of scenarios with common-mode or correlated pathways may lead to criticality of a number of packages over time with a probability that may or may not be much less than that of a single criticality in a single package. Once a certain scenario or pathway is established, criticalities in other similar packages

by that pathway or a closely correlated one are not statistically independent. Therefore, in such cases one may need to consider the consequences (and probabilities) of more than one criticality event under this criterion.

The criticality consequence criterion refers to a single criticality to provide a screening criterion for individual waste forms. This is a very coarse criterion intended to identify those design/waste form combinations that can immediately be characterized as requiring a detailed Total System Performance Assessment (TSPA) evaluation. It is expected that most design/waste form combinations will pass this screening. However, criticality consequences may subsequently be identified with unacceptably large contribution to the overall dose at the accessible environment (the ultimate consequence, determined by the TSPA). This feed of all criticality consequence results to the overall TSPA will be added to the Overview (Figure 1-1) when the Topical Report is revised.

With respect to the issue of consequences from several criticality events, the comprehensive criticality consequence evaluation performed in connection with TSPA will include consideration of multiple criticality events, including those from common mode failures. It should be noted, however, that a major source of common mode failure is either design or manufacturing defect. DOE expects to be able to show that the probability of such a defect is very small, and even if such a defect were to occur it would not lead to a criticality.

1-3 Clarify the range of applicability of the methodology discussed in Item G.2, page 1-5.

The footnote on page 1-1 states that the methodology and processes are to be applicable to all different waste forms (WFs). Item G.2 is inconsistent with this in requesting consideration of the validation process for a limited class of waste forms (i.e., commercial Spent Nuclear Fuel (SNF)).

As stated in the Topical Report, the planned range of applicability for the methodology is any waste form. However, some elements of the methodology, such as Item G.2, are applicable to specific waste forms. This is discussed in the first paragraph on the top of p. 1-4. DOE recognizes that applicability to any given waste form will need to be demonstrated. Item G.2 was intended only for commercial Spent Nuclear Fuel. DOE plans to discuss the requirements that will make up the isotopic model validation process for specific waste forms in separate addenda (in the case of naval fuel) or validation reports.

DOE will modify Item G as follows:

"G. The criticality model validation process described in Section 4.1.3 is acceptable in general for model validation. Specifically, the process presented in Subsection 4.1.3.2 for calculating the CL values and the process presented in Subsection 4.1.3.3 for establishing the range of applicability of the CL values define the validation process for the

criticality model. This validation process will be followed to calculate CL values for specific waste forms and waste packages as a function of the degradation conditions. Acceptance of CL values and their applicability for postclosure repository conditions will be sought in validation reports and referenced in the License Application."

1-4 Explain why a broader range of configurations is not discussed in Item J.1, page 1-5.

The configuration identification process may fail to identify those configurations with the greatest potential consequences, i.e., configurations with potentially positive feedback. Such configurations should be identified using a process, supplementing the existing proposed method, whereby the most significant credible or postulated configurations are first identified and then either eliminated or further considered based on an evaluation of the probabilities of mechanisms that could produce such configurations.

A "broader range" of configurations was not discussed in J.1 because the range discussed already covers the full spectrum of recognized critical configurations. The process for identifying configurations will trace all recognized movements of fissionable material and therefore will fully address credible configurations with the greatest potential consequences. The range of configurations considered includes criticalities inside and outside a waste package (i.e., all locations), and both transient and steady-state events (i.e., all types). "Configurations with potentially positive feedback" are a subset of the generic transient critical configurations.

The methodology already has the steps asked for in the RAI. Items B and C (p. 1-4) describe the process for identifying the credible configurations classes from the standard scenarios. Item C (p. 1-4) covers using the potential mechanistic processes to determine which configuration classes can exist, and Item E (p. 1-5) covers determining the probability of the potentially critical configuration classes.

The Topical Report does not fully specify the parameter ranges for the standard scenarios or the resulting configuration classes. This is part of the application of the methodology. Nevertheless, it is the DOEs intention that they be comprehensive, and include configurations of the kind suggested by this RAI. For example, DOE intends to evaluate the probability of all configurations that have been identified as potentially autocatalytic in published articles.

Section 1.3 Scope

1-5 Explain why the scope of the TR does not correspond to the methodology and processes actually described in the TR.

The methodology and processes discussed in the Topical Report were developed

primarily with commercial Light Water Reactor (LWR) fuel in mind. In addition, the LWR discussions are mostly restricted to Pressure Water Reactors (PWRs), with little consideration of the Boiling Water Reactors (BWRs). The staff expects to see numerous exceptions and differences in the methodologies ultimately used for naval Spent Nuclear Fuel (SNF), other U.S. Department of Energy (DOE) SNF, other highly-enriched materials, graphite-moderated fuel, and vitrified High-Level Waste (HLW). The acceptability of the methodology described in the Topical Report to the broad variety of waste forms (e.g., all 250 DOE SNF types) cannot be established in this review.

The scope of the Topical Report is primarily to describe a methodology for predicting the potential for, and the consequence of, criticality during the postclosure period of the geologic repository. The methodology described in Section 3 of the Topical Report is intended to be a process to be applied to all types of SNF and HLW expected at the repository. The methodology is based on specific criticality and isotopic neutronic models coupled to degradation models. Section 4 discusses the specific models for Light Water Reactor (LWR) fuel for which general NRC acceptance was desired. PWR fuel data are used throughout the Topical Report as an example spent fuel type to illustrate application of the generalized methodology. The examples and waste form specific items will be removed from the main body of the Topical Report, or more clearly labeled, when it is revised. Once this is done, the scope in Section 1.3 will be consistent with the remainder of the report.

It is expected that differences in the details of specific methodology components will occur with application to different spent fuel types. DOE recognizes that applicability to any specific waste form will need to be demonstrated. Addenda to the Topical Report will describe in detail the specific information necessary for application of the criticality analysis methodology to each waste form type (e.g., DOE SNF including naval SNF or HLW). DOE plans to provide validation of the methodology components for each of the waste forms, with the exception of naval SNF, in validation reports. The validation of the methodology for naval SNF is expected to be contained in the classified Addendum to the Topical Report for naval SNF. The planned workscope for a commercial SNF validation report is given in Attachment A to this response.

Details of the application of specific models to any of the spent fuel types is beyond the scope of the Topical Report at this time. DOE plans to describe application of the specific models to the actual design in the License Application or its references.

Section 1.4 Quality Assurance

1-6 Clarify the statement that "the information presented in this topical report is not design information that can be used to support procurement, fabrication, or construction." with respect to Quality Assurance.

If the methodology is formulated (e.g., specifying critical limit, dismissing configurations not having potential for criticality etc.) based on the data

presented in this report, and the design of the criticality control systems in the waste packages (WPs) is based on this methodology, it is not clear how these data are not used, directly or indirectly, in the design of the waste package. In particular, some of the references (e.g., CRWMS M&O 1998e) state that "this document will not directly support any DOE Office of Civilian Radioactive Waste Management (OCRWM) construction, fabrication, or procurement activity and therefore is not required to be procedurally controlled as to be verified (TBV)."

Development of the Topical Report was subject to DOE Office of Civilian Radioactive Waste Management (OCRWM) Quality Assurance Requirements Description (QARD) controls (DOE/RW-0333P). The purpose of the statement in question was to note that the methodology documented in the Topical Report can not be used directly for procurement, fabrication, or construction activities in accordance with the procedures implementing the OCRWM QARD. The procedures implementing the QARD controls require design information for those activities to be controlled in drawings and specifications supported by design analyses. The Topical Report provides an analysis methodology, but does not provide drawings or specifications, and so cannot be used to (directly) support procurement, fabrication, or construction. The Topical Report and its internally developed supporting calculations, analyses, and technical reports were prepared in accordance with the procedures implementing the QARD requirements. The use of the methodology from the Topical Report in design analyses supporting drawings or specifications would be acceptable from the standpoint of the procedures implementing the QARD. The methodology in the Topical Report, when accepted, will be used in design analyses which will be used to generate drawings and specifications which in turn will be used to support procurement, fabrication, and construction. The statement in the QA Section of the Topical Report was not intended to imply a quality assurance deficiency, but rather to reflect limitations on use of a methodology document in accordance with procedures.

1-7 Specify what part of the Actinide-Only Burnup Credit Topical report is used in this topical report.

The "Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Packages (DOE 1997)" has not been approved by the NRC Spent Fuel Project Office.

This section of the topical report is required per the OCRWM procedures to note interfaces issues with the development of this report. The burnup credit issue was considered an internal OCRWM interface issue since two organizations of OCRWM (transportation and disposal) were addressing burnup credit issues with the NRC.

As noted in the topical report (last sentence in the third paragraph of Section 1.4), some of the data and parts of the methodology from the *Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Package* are referenced in the *Disposal Criticality Analysis Methodology Topical Report*. The data "from" the Actinide-Only Topical Report referenced in the *Disposal Criticality Analysis Methodology Topical*

Report include the chemical assay data and the benchmark criticals (laboratory critical experiments). The data actually come from second- and third-tier references which both reports reference. Because this reference is so indirect and not needed to support the Topical Report, DOE will remove the statements that reference the Actinide-Only Topical Report.

Section 1.5 Overview of the Methodology

1-8 Justify the approach used to dismiss configuration classes as having the potential for criticality based on an evaluation of a given configuration with parameter values at some selected points for each of the configurations (Figure 1-1).

If configuration classes are dismissed based on evaluation of a given configuration with parameter values at several points rather than examination of full range of parameter values, it is possible to dismiss a configuration or even configuration class, which has potential for criticality. It seems it would be more appropriate to perform the criticality analyses for a full range of parameter value for each configuration, examine those results against the Critical Limit criterion, and then decide if that particular configuration or configuration class is acceptable for disposal from a criticality standpoint.

Figure 1-1 requires that configurations be evaluated "for a range of parameters and parameter values." This is from the box above the "satisfy Critical Limit (CL) criterion" decision diamond. The intent of the box above the first decision point in Figure 1-1 is to evaluate the range of parameter values so as to not let a potentially critical configuration be overlooked. In practice this may include examining only bounding values for certain key parameters and the range of values for other parameters. For example, a configuration class of intact commercial fuel in a waste package may be evaluated with 5 wt% U-235 fuel. If the case satisfied the CL criterion, there would be no need to run the same configuration with 4 wt% U-235 fuel and the same or greater burnup. But if the CL criterion was not satisfied, the configuration class would be considered potentially critical, and additional cases would be run to develop regression expressions (the second box to the right of the satisfy CL criterion decision diamond). In any case, the justification for acceptance of a configuration will be provided in the supporting documents for the License Application to allow NRC review and concurrence.

The referenced box in Figure 1-1 will be modified as follows:

> "Perform Criticality Analysis (k_{eff}) of Defined Configurations (for each class) **Over the** Range of Parameters and Parameter Values"

As part of the NRC staff's review of the supporting documents for the License Application, DOE assumes that the staff will review the documents, developed based on the methodology from the Topical Report, to verify that the range of parameters and

parameter values evaluated in the cases is sufficient to justify dismissing configuration classes.

Chapter 2.0 Regulatory Perspective

2-1 Throughout this section, there are references to Regulatory Guide 3.58. However, it, as well as 3.1, 3.4, 3.43, 3.45, 3.47, 3.57, 3.68, 3.70, and 8.12, has been superseded by Regulatory Guide 3.71, published in August 1998. The references should either be updated or an explanation for the choice to use Regulatory Guide 3.58 should be provided.

The DOE will revise Section 2.3.3 of the Topical Report to replace references to Regulatory Guides 3.4 and 3.58 with reference to Regulatory Guide 3.71 as recommended in the RAI. The section will note that Regulatory Guide 3.71 endorses use of ANS/ANS-8.1-1983, ANSI/ANS-8.10-1983, ANSI/ANS-8.15-1981, and ANSI/ANS-8.17-1984, with certain caveats and exceptions discussed in the Regulatory Guide. The section will then refer to DOE's commitment to these standards as stated in the preceding Topical Report section and will describe DOE's commitment to the Regulatory Guide. Neither the Regulatory Guide nor the standards referenced in it are explicitly applicable to disposal, and some of the referenced standards are clearly inapplicable to the postclosure period. The exceptions DOE proposes to take as noted in the Topical Report will remain. In addition, the revised Topical Report section will note that the DOE believes the remaining standards referenced in the Regulatory Guide are inapplicable to postclosure disposal, and therefore commitment to them via this topical report is inappropriate.

In addition, Chapter 2.0 will be revised to reflect the NRC's new Yucca Mountain regulations at 10 CFR Part 63, if these regulations are issued by the time the Topical Report is revised to address the RAI comments.

Chapter 3.0 Methodology

Section 3.1 Standard Criticality Scenarios

3-1 Indicate how the effects of disruptive events will be considered in the evaluation of potential criticality events in the repository.

The scenarios listed in Figure 3-1 and 3-2 appear to be comprehensive for an undisturbed repository. However it is not clear whether the potential effects of disruptive events have been adequately considered. Failure to consider all potential scenarios that could result in a criticality event could result in an underestimation in the probability of a critical event occurring within the repository. Some potential effects of disruptive events include:

(a) *Seismic events could cause the waste packages to rotate on their invert, potentially allowing corrosion products to be released while the hole in the waste package is facing down. Later, the hole could rotate back to the top of the package, allowing the package to fill with water.*

(b) *A volcanic event, although a low probability event, could fail many waste packages and force their contents into a compact configuration encased in lava at the end of the tunnel.*

The Master Scenario List shown on Figures 3-1 and 3-2 is utilized to determine configuration classes and the probability of each class. Within each class multiple configuration classes are possible. Sections 3.2 and 3.3 describe how the configurations are determined in each class. Disruptive events are analyzed as part of the configuration determination. In Section 3.2, following the list of 10 steps in the geochemical processes, is a list of physical processes. The fourth item in that list specifies that considerations of external events such as seismic activity are evaluated at appropriate intervals in the process. DOE plans to evaluate possible configurations that can be generated by disruptive events as is mentioned in Section 3.2 of the Topical Report.

DOE believes that the potential effect items a) and b) from the RAI are application issues and therefore are not within the scope of the Topical Report. These issues will be fully addressed in the supporting documents for the License Application. The following discussion is provided for informational purposes only. The information provided is preliminary and will be updated as necessary with additional evaluations or experiments prior to the License Application.

a) DOE expects to show that the seismic event ground motion at a 200 meter depth would not be sufficient to cause a significant rotation of a horizontally emplaced waste package. DOE also expects to show that by the time of significant waste package degradation there would likely be significant degradation of the drip shield. This, in turn, would be expected to transfer significant pressure from the backfill directly onto the entire circumference of the waste package, thereby greatly inhibiting any rotation. DOE plans to complete all the supporting calculations for the final EBS design in time for License Application.

b) DOE expects the effect of magmatic intrusion would be to disperse the contents of a waste package. However, DOE plans to evaluate the probability of an occurrence such as specified in this RAI comment , and DOE plans to evaluatethe k_{eff} for the credible range of disruptive event parameters with the application of the methodology for the License Application.

The configuration classes that are developed based on disruptive events are planned to be discussed in supporting documents for the License Application.

3-2 Justify the scenario selection process used to focus on the degraded modes and phenomena that produce the critical configurations of interest.

The process as now proposed may not identify and address some scenarios that should be considered. For example:

(a) The internal scenarios do not give adequate consideration to criticality at the less-burned ends of the fuel. Even in scenarios where the basket poison material, or absence thereof, is uniformly distributed over the length of the active fuel material, criticality will occur predominantly at the fuel ends. End-effect criticality is made even more relevant by scenarios where fuel and basket poison material are displaced axially relative to one another. Such displacement or shifting would remove poison from where it is most needed, i.e., at the ends.

(b) For external near-field and far-field criticality, more focused consideration should be given to scenarios with potentially positive neutronic feedback characteristics. This could be done by first identifying hypothetical configurations that produce positive feedback effects and then evaluating the credibility or likelihood of mechanisms that might form such configurations. The methodology as now proposed does not seek and address those criticality scenarios that have the greatest potential consequences. Furthermore, the probability criterion for events with potentially high consequences should be lower than that for events with less consequences.

The scenario selection process does not focus on generating critical configurations of interest. Rather, the process evaluates what scenarios are mechanistically possible for a given waste form, waste package, engineered barrier, and repository design system. The process will provide input to the processes described in the Topical Report that will evaluate potentially critical configurations.

The example configurations a) and b) noted in the RAI are application issues and therefore are not part of the methodology or the Topical Report. The following discussion is provided for informational purposes only. The information provided is preliminary and will be updated as necessary with additional evaluations or experiments prior to the License Application.

a) Axial effects for commercial SNF fuel are planned to be included in the neutronic models. The consideration of axial or "end-effects" is noted in Section 4.1.3.1.4 (2nd paragraph, 2nd sentence) of the Topical Report in the requirements of the isotopic model. The example with BWR fuel in Appendix C of the Topical Report included accounting for the axial effect with 10 axial nodes (Section 3.1.1). The exact number of nodes to be used will be justified in the commercial SNF validation reports. The detailed description of how a feature of one waste form (axial effects for commercial SNF) will be modeled is an application issue. DOE plans to revise the Topical Report to state that the criticality model needs to account for specific waste form features like axial burnup in commercial SNF.

With respect to absorber plate displacement for commercial SNF, there is likely to be no room to displace the borated stainless steel plates axially realative to the fuel in the current designs. The absorber plates extend essentially the full length of the fuel assembly cavity in a waste package. The length of the cavity that the plates do not extend to is less than the length of an end plate of a fuel assembly. Configurations involving relative movement between waste form and absorber are covered as part of degradation analysis. For designs that contain control rods, DOE plans to consider a mechanism for holding the control rods in place to prevent displacement. Again, DOE plans to consider degraded configurations involving relative movement between control rods and the waste form in the methodology. The only cases where significant separation of the control rods and waste form could occur are when the waste form has degraded. The preliminary result of thoses example calculations referenced in Appendix C of the Topical Report (Section 3.1.4), indicate the degraded commercial SNF cases are less reactive than the intact fuel cases, and the control rods may not necessarily be needed then.

The displacement cases do not appear to be of much concern for criticality. Cases that are more likely to have potential for criticality are configurations where preferential corrosion occurs in the plates at one end of a waste package and cases where there are problems with construction of a waste package (i.e., use of a non-borated plate in a section). The first case will be addressed as part of the degraded analysis, the second case will be addressed in an extension of the early (juvenile) failure analyses. Complete evaluations are planned to be provided in supporting documents for the License Application.

b) As stated in the response to RAI 1-4, and in Section 4.4.1.2 of the Topical Report, all credible critical configurations with positive feedback effects (and those without) will be evaluated. The last sentence of this RAI item suggests the need for a lower probability threshold for events with a greater consequence. The probability threshold is not intended to exclude low-probability events from further consideration so it is not analogous to probability thresholds applied to preclosure safety analyses. The concept of risk, which is the product of criticality probability multiplied by a measure of the criticality consequence, is designed to accomplish this compensation (of probability against consequence). The licensing argument will be based entirely on the measure of risk, consistent with the requirements of 10CFR63. Therefore DOE believes there is no need for a probability criterion that varies with consequence.

3-3 Explain why a discussion of the fast-fissionable, non-fissile actinides that by themselves can sustain a critical chain reaction is not included in this section.

The discussion of fast criticality scenarios in which little or no moderation is required should be extended to include "minor actinides" (see ANSI/ANS 8.15) that by themselves can sustain a critical chain reaction with fast neutrons only. Such actinides are sometimes called "fissible." The TR should indicate how these actinides have been considered with regard to their abundance over time in various waste forms and should discuss the bases for any conclusions about their

significance (or lack thereof) to repository criticality.

A discussion of criticality scenarios that include "minor actinides" is already included in the methodology as described in Section 3.1 of the Topical Report. The methodology addresses "fissionable" isotopes, which covers the minor actinides of concern in ANSI/ANS-8.15. The examples in the Topical Report, which do not specifically address the minor actinides, are not intended to represent the complete application of the methodology. DOE plans to address specific configurations with the minor actinides which are fast-fissionable, non-fissile in the supporting documents for the License Application.

DOE believes that discussions of the specific scenarios as suggested in the RAI are application issues and therefore are not part of the methodology or appropriate for inclusion in the Topical Report. The following discussion is preliminary and is provided for informational purposes only.

The minor actinides are not expected to be of concern. These actinides have been considered in the past and dismissed as having too low an abundance to sustain a criticality, and it is considered incredible that they could accumulate in significant quantities (Gore et al. 1981, Brookins 1978, Allen 1978).

With earliest breach of the waste package expected to occur in the 50,000-year to 100,000-year time frame, no minor actinides, other than Np-237, are expected to be present in significant quantities for transport, as is illustrated in Table 3.3-1 below. In addition, their decay products do not affect the inventories of fissile isotopes present to which they decayed. Oak Ridge National Laboratory (Allen 1978) observed that neptunium oxide and americium oxide have such large minimum critical masses that they probably do not present a potential criticality problem.

Table 3.3-1 Minor Actinide Isotopes

Isotope	Half-life (GE 1989)	Critical Mass (kg)[a]	Fraction Present [b]
Np-237	2.14E6 years	45	0.0201
Am-241	432.7 years	-	-
Am-242m	141 years	-	-
Am-243	7.37E3 years	78.9	0.0002
Cm-242	162.8 days	-	-
Cm-243	29.1 years	-	-
Cm-244	18.1 years	-	-
Cm-245	8.5E3 years	3.03 [c]	0.00002 [c]
Cm-246	4.76E3 years	3.03 [c]	0.00002 [c]

a Critical masses are for a system moderated and reflected by granite at 10,000 years (Allen 1978)

b The fraction of a critical mass present from a PWR assembly burned to 33 GWd/mtU, at 10,000 Years (assuming granite moderated and reflected cases) (Allen 1978)

c Values from Allen 1978 are on an elemental basis, values shown are for a combination of Cm-245 and C-246

\- Indicates that essentially all the isotope has decayed by 10,000 years.

Gore, B. F. , U. P. Jenquin, and R. J. Serne, (Gore et al.) 1981. *Factors Affecting Criticality for Spent Fuel Materials in a Geologic Setting*. PNL-3791. Richland, Washington: Pacific Northwest Laboratory, April 1981.

Allen, E. J. 1978. *Criticality Analysis of Aggregations of Actinides from Commercial Nuclear Waste in Geological Storage*. ORNL/TM-6458. Oak Ridge, Tennesse: Oak Ridge National Laboratory, August 1978 pgs 34, 26, 30.

Brookins, D. G. 1978. *Geochemical Constraints on Accumulation of Actinide Critical Mases from Stored Nuclear Waste in Natural Rock Repositories*. ONWI-17. Albuquerque, New Mexico: Office of Nuclear Waste Isolation, Battelle Memorial Institute, December 1978.

GE (General Electric) 1989. *Nuclides and Isotopes, Fourteenth Edition, Chart of the Nuclides*. San Jose, California: General Electric Company, Nuclear Energy Operations, 1989.

Section 3.1.2 External Scenarios

3-4 Confirm that far-field configuration classes FF-3c, 3d, and 3e are located in the saturated zone.

The text in item 1 of Section 3.1.2 contradicts figure 3-2b in assigning these configurations to the unsaturated zone. The distinction is important in modeling hydrologic and geochemical processes.

FF-3c could be in either saturated or unsaturated zones. FF-3d must be in the saturated zone by definition, and FF-3e implies spending some time in the saturated zone in order to reach the Franklin Lake Playa. Item 1 of Section 3.1.2 was not intended to exclude the saturated zone. It will be reworded as follows:

"1. Accumulation, by chemical reduction, of fissionable material by a mass of organic material (reducing zone) located beneath the repository, at a narrowing of the tuff aquifer, or at the surface outfall of the saturated zone flow (FF-3c, 3d, 3e, respectively). The combined probability of the existence of such a reducing zone and its being encountered by a flow bearing fissionable material is extremely low (CRWMS M&O 1998i)."

3-5 Explain why, in item 3, configuration NF-1b includes only a reducing reaction with tuff as a mechanism for precipitation of fissile solutes in the near-field below the waste package.

Other chemical reactions should be considered as causing such precipitation, such as changes in aqueous chemistry related to the presence of concrete and tuff. This comment reflects the desire for completeness in modeling the configurations.

The language in the Topical Report identifying reducing reactions as the potential source of external accumulation was only intended to represent the typical possibilities. The evaluation of the accumulation from the waste package outflow (source term) will be accomplished by geochemical-transport computational methods that consider all recognized chemical reactions, as discussed in Sections 4.2.2 and 4.2.3 and in the response to RAI 4-32. Item 3 will be modified as follows to reflect that these are illustrative only:

> "...other void space of the near-field, obtained from processes such as adsorption or from a reducing reaction ..."

The following discussion illustrates the additional types of accumulation reactions and mechanisms that will be considered for the License Application. The discussion is provided for informational purposes only. The information provided is preliminary and will be updated as necessary with additional evaluations or experiments prior to submittal of the License Application. DOE plans to evaluate accumulation by adsorption . This capability is not available in EQ3/6, so DOE plans to use the geochemistry-transport code PHREEQC.

The principal non-reduction mechanism of deposition appears to involve destabilization of aqueous uranium (U) and plutonium (Pu) carbonate complexes, due to lowering of the dissolved $CO_3^{=}$; the latter can occur through reaction with calcium (Ca)-silicates in tuff to create calcite, or by lowering of the system pH against the constraint of fixed CO_2 fugacity. A preliminary analysis of external accumulation from waste packages containing mixed oxide wastes showed deposition of U was greatest (~20 kg) when pH 4 solutions were reacted with crushed tuff invert. Such a low initial pH is extremely unlikely, but could result from oxidation of chromium in the stainless steels. To achieve such a low pH, the corrosion of the stainless steel would have to be so fast that it would all be corroded in 500 years. The likelihood of significant external accumulation by this mechanism is further reduced by the fact that subsequent, neutral pH solutions would be expected to redissolve a significant fraction of any actinide accumulation.

Calculations (CRWMS M&O 1998) for co-disposal waste packages (those containing both SNF and HLW) showed that significant external accumulation of fissile material requires: (1) in the waste package, extremely rapid degradation of the fissile-containing waste form, along with reasonably rapid degradation of the glass, to produce and alkaline-carbonate solution capable of mobilizing the actinides and (2) in the drift, reaction of the effluent with Ca-containing silicates, which changes the effluent chemistry enough to induce precipitation. So far this model yields significant (sub-critical) precipitation only for Pu, and only insignificant accumulations of U. Even when the effluent encounters incompletely oxidized corrosion products (e.g. Fe_3O_4 from degraded steel sets), which have the capability to support reducing reactions, it is difficult to achieve significant reductive precipitation before the effluent passes out of the invert, or all the material with reductive capability is oxidized by air.

In the accumulation reactions evaluated thus far, the reducing mechanism is the most effective for producing Pu accumulations. One extreme reducing scenario, in which the effluent from the waste package was assumed to encounter plates of unaltered carbon steel, produced up to ~10 kg Pu deposition, spread over a substantial volume below the waste package. However, it is considered unlikely that substantial, unreacted steel fragments could be maintained up to the period of waste package breach; preliminary in-process calculations suggest the steel drift supports would be heavily corroded within ~1000 years, well before the probable time of waste package breach.

DOE is extending the prior work to consider: (1) a wider range of deposition environments, with more realistic models for fO_2 (oxygen fugacity, idealized partial pressure) control by O_2 diffusion and effluent reaction with Ca-silicates and reduced phases in the tuff, and reaction with partially-corroded (magnetite-rich) steel fragments; (2) greater analysis of U silicates as possible precipitants; and (3) tighter coupling between the evolution of the waste package effluent chemistry and the invert and drift materials. The tighter coupling of waste package effluents and ex-package materials will likely decrease the calculated deposition, since the previous analysis picked snapshot waste package solution composition (with high dissolved actinide content); in fully coupled models, such solutions will be "chased" by fluids that favor re-dissolution of the actinides.

It should be noted that the new design concept for the Engineered Barrier System (called EDA-II) has very little concrete in the drift liner or invert material. Ground support in this design may include some concrete, but the total amount of concrete in the drifts will be much smaller than in previous designs. DOE plans analyses that will appropriately account for this concrete as well as the possibility that concrete in the design, but outside the emplacement drift, could impact potential criticalities.

CRWMS M&O 1998. *Report on Intact and Degraded Criticality for Selected Plutonium Waste Forms in a Geologic Repository, Volume II: Immobilized In Ceramic.* BBA000000-01717-5705-00020 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19981007.0180.

3-6 Clarify whether credit will be taken in the criticality analyses for the assumption that there is no mechanism for completely sealing the fractures in the bottom of the drift so any in-drift accumulations of water will only be present for a few weeks.

Previous investigations have indicated that thermal alteration of the rock surrounding the repository or microbial growth has the potential to seal fractures, at least in local portions of the repository (Lin and Daily, 1989[1]). Although it appears that the scenarios listed in figure 3-2 include the potential for water to pond on the bottom of the drift, if credit is taken for the short duration of ponding water in the drift, this assertion that fractures cannot become sealed will have to be justified.

The DOE believes that the specific inputs or assumptions used in an analysis are an application issue and not appropriate for inclusion in the Topical Report. The methodology requires taking into account expected and potential conditions in the repository. Justification for the conditions used in evaluations will be provided in the evaluations or supporting documents for the License Application. While preliminary evaluations have indicated short-duration ponding as noted in Topical Report Section 3.1.2, Item 4, the final conditions that will be used for the License Application have not been determined. DOE plans to describe and justify these final candidates in the supporting documents for License Application.

Item 4 will be changed to the following to clarify that the ponding discussion is an example:

> "4. Accumulation of fissionable material in a standing water pond in the drift, configuration NF-4a, reached from scenario E. This scenario involves waste packages that may not have been directly subjected to dripping water, but are located in a local depression so that water flowing from other dripping sites may collect around the bottom of the package during periods of high flow. A variant of this configuration class could have the intact, or nearly intact, waste form in a pond in the drift (configuration NF-5a). Such a configuration would be evaluated for waste forms that could be demonstrated to be more robust with respect to aqueous corrosion than the waste package. The detailed analyses for the License Application will evaluate the probability of occurrence for such a pond, of sufficient depth to cover most of the assemblies, while the assemblies are stacked in a geometry favorable to critcality."

The DOE believes that a discussion of the specific ponding assumptions used in the criticality analysis mentioned in the RAI is an application issue and therefore is not appropriate for inclusion in the Topical Report. The following discussion is provided for informational purposes only. The information provided is preliminary and will be updated as necessary with additional evaluations or experiments prior to the License Application.

Based on current work summarized below, DOE expects to be able to show that ponding in the drift to a depth of more than a few centimeters is incredible. DOE is presently investigating the possible mechanisms for plugging the fractures (deposition of minerals, clay buildup, deposits from microbial growth); all indications thus far are that they would be very unlikely to fill all the fracture openings in the drift floor beneath the waste package. In the meantime DOE has performed a simple conservative analysis of the maximum depth that could be sustained under the worst-case conditions that all the fractures are plugged over a certain length of drift (or there happens to be a section of drift that has no fractures in the floor to begin with).

A variant of the pond concept is a groundwater mound inside the invert, which is supported by the flow resistance of a fine-grained backfill and initiated by a large water

pulse. It is expected that the DOE Yucca Mountain hydrologic modeling effort will soon produce a realistic value for the largest water pulse. In the meantime, the current worst case estimate of 66 m^3 in one week (1.09E-4 m^3/s) is used. This value corresponds to an infiltration rate of 165 mm/y focused in a week and from an area of 400 m^2. It is assumed to be released in the drift over a localized area of 1 m^2. The mound height is maintained by the porous media resistance to lateral outflow from this source, according to the following formula for flow in a vertical plane through the drift axis.

$$\frac{Q}{4K}L = \int_0^{h0} A(h)dh,$$

where Q is a line source perpendicular to the plane of the flow, K is the invert conductivity, h is the height of water above the bottom of the drift (ponding height), h_0 is the peak height at the center of the mound, and L the maximum horizontal extent of the mound. *A(h)* is the cross sectional area of the mound perpendicular to the axis of the drift:

$$A(h) = 0.5r^2(\phi - \sin\phi) \; with \; \phi = 2\left(a\cos\left(\frac{r-h}{h}\right)\right)$$

where r us the radius of the drift. Assuming it has a conductivity of 0.1 cm/s (reasonable for fine sand of 0.5 mm grain size), and a spacing (L) of 24.6 meters between unplugged fractures (which would have a probability of 1.3E-5 of occurring naturally, according to measured fracture spacing distribution data for the drifts drilled thus far), a mound can form to a high of 59 cm above the drift floor.

Ponding by this mechanism will be of short duration, because of the episodic flooding necessary for the mound behavior to reach significant depth. The one-week duration in the present example would not be long enough for any significant radionuclide inventory increment in a steady-state criticality. Considered as a transient criticality, the extremely unlikely one week water pulse of this example would still provide only the very slow reactivity insertion rate considered in the example of Appendix C of the Topical Report. That slow insertion rate transient was shown in Appendix C to produce no significant pressure or temperature pulse.

The profile of the water height as a function of the distance along the drift axis is shown in Figure 3.6-1.

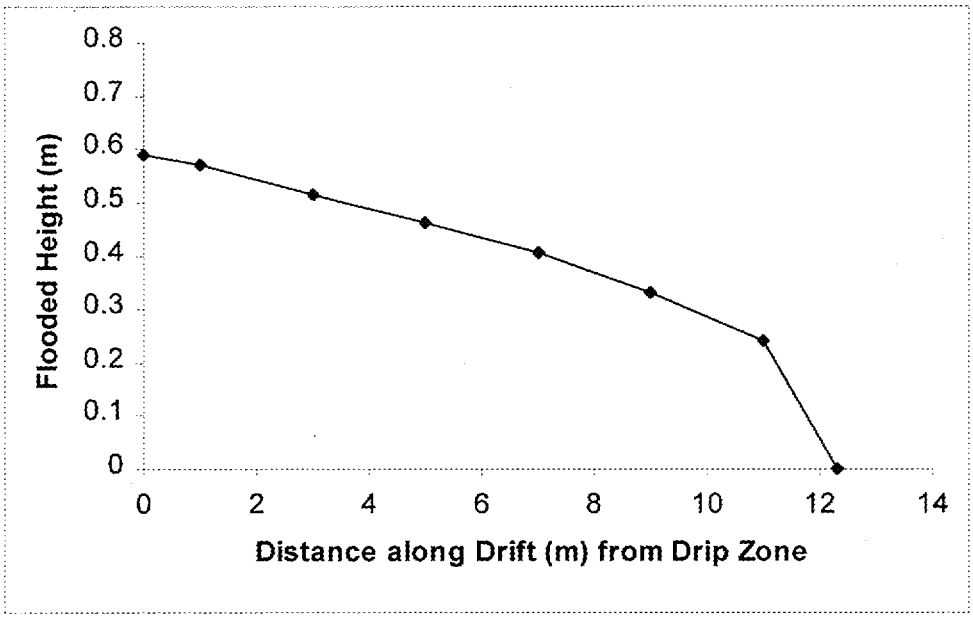

Figure 3.6-1 Saturated Zone Profile (0.5 mm sand, episodic flood)

3-7 Explain why, in item 3, configuration NF-1b includes only a reducing reaction with tuff as a mechanism for precipitation of fissile solutes in the near-field below the waste package.

Other chemical reactions should be considered as causing such precipitation, such as changes in aqueous chemistry related to the presence of concrete and tuff. This comment reflects the desire for completeness in modeling the configurations.

See response to RAI 3-5.

3-8 Correct item 5 to state that that the final two configurations are NF-3b and 3c, rather than FF-3b and 3c.

The context of the sentence implies incorrectly that the listed sites of colloidal accumulation are all in the far field. In addition, in the final sentence "open fractures" should be specified as being in concrete to be consistent with Figure 3-2a. These changes will correct the impression from item 5 that all colloidal accumulation sites are far-field.

The designation NF (for Near Field) was inadvertently omitted for 3b and 3c and will be corrected in a revision to the Topical Report. The final sentence of item 5 in 3.1.2 will be modified to indicate that the near-field open fracture accumulation would be in concrete and to indicate that there can be open fracture accumulation in the far field, as follows:

> "Such transport and accumulation could lead to the far-field configurations FF-2a, 2b, 2c, for final accumulation in dead-end fractures, clay or zeolites, and topographic lows. It could also lead to the near-field configurations NF-3b, 3c,

for final accumulation in the invert in open fractures of solid material or porespace of granular material, respectively."

Figure 3-2b Part 2 will also be modified to indicate this latter fact (by adding to the caption of FF-2a), as follows:

"Filtration and concentration of colloids in granular pore space".

Section 3.2 Determining Internal Configurations

3-9 State whether temperature is included among the parameters quantified at this stage of the methodology and describe possible thermal variations.

Because equilibrium states and degradation/reaction rates for WP internal components (including WF) are temperature-dependent, all geochemical modeling should include sensitivity to temperature variations, including those caused by repository heating and cooling and by criticality events. With respect to internal configurations, the example analysis of appendix C refers to EQ6 calculations described in CRWM M&O (1998e[2], appendix C, reference list). The discussion in this reference does not explicitly mention temperature constraints on models. Recent DOE modeling of the near-field (Harbin, 1998[3]) predicts that temperatures of close to 100 ° C may persist at the repository horizon 5000 yr after closure. A more recent repository design, EA-11[4] yields lower drift temperatures, but the waste package would still experience temperatures above 60 ° C for at least 2000 yr. Thermal variations have a strong effect on degradation processes and rates for Waste package internal components. Furthermore, high temperatures would affect water chemistry (e.g., see composition of J-13 equillibrated with tuff at 90 ° C in Wronkiewicz et al. (1992)[6]} This comment applies also to discussions of internal and external geochemistry models in topical report sections 3.3, 4.2.2, and 4.2.3.

The proposed methodology has the capability to evaluate the temperature sensitivity of geochemistry effects in the computational method (e.g., both EQ3/6 and PHREEQC). Effects of thermal variation effects on the external and internal geometry models will be documented in supporting reports for the Licensing Application.

DOE believes that a discussion of the specific parameters used in the criticality analysis mentioned in the RAI is an application issue and therefore is not approapriate for inclusion in the Topical report. The following discussion is provided for informational purposes only. The information provided is preliminary and will be updated as necessary with additional evaluations or experiments prior to the License Application.

DOE does not expect elevated temperature to play a very significant role in waste package chemistry for the following reasons: (1) Many reactions of interest, which control mobility and precipitation of actinides, show a weak or retrograde temperature

dependence in experiments; (2) Below 50°C significant accumulation of water in the waste packages may be impossible, because the rate of evaporation exceeds the drip rate; (3) In the Viability Assessment (DOE 1998) models, the first probable waste package breach is likely to occur only after the average temperature is below 50C; and (4) In the latest repository design, ventilation will be used to reduce temperature. The following paragraphs provide evidence for the first assertion. The other three assertions are simply observations of the natural phenomenon in the repository, predicted behavior of the waste package design, and a feature of the latest repository design. DOE plans to justify the assertions in the supporting documents for License Application.

Wilson and Bruton (1989; relevant table is Table 3, p 10) found the temperature-dependence for uranium solubility was weak, and that plutonium solubility actually decreased with temperature. Similar results for plutonium were recently found by Efurd et al. (1998, pp. 3893-3900). Thus it is likely that an increase in temperature may lower the actinide concentrations leaving the waste packages.

Wruck and Palmer of Lawrence Livermore National Laboratory (1997) proposed a method to extend the current database to include higher-temperature extrapolation for actinide complexes and solids; this method will be considered in upcoming months, but the addition of higher-temperature data will follow development of a QA framework consistent with DOE's and NRC's (NUREG 1298) definitions of "accepted" and "qualified" data.

For calculations of internal criticality, existence of retrograde solubility typically means that use of 25C data is conservative. In systems with sufficient phosphate, the projected solubility-limiting phases for gadolinium is $GdPO_4 \cdot H_2O$, which has a weak retrograde solubility (Firsching and Brune 1991). In low-phosphate systems, the projected solubility-limiting phase is $GdOHCO_3$ or $Gd_2(CO_3)_3$. The temperature dependence for the gadolinium carbonates has not been determined in experiments, but most carbonates are retrograde. Thus a higher-temperature package may increase the likelihood that the criticality control material will remain with the fissile materials.

DOE plans to include the information dicussed in this response in the validation report for the degradation/geochemistry models.

DOE 1998. *Viability Assessment of a Repository at Yucca Mountain – Total System Performance Assessment – Volume 3.* DOE/RW-0508/V3. North Las Vegas, Nevada: DOE. ACC: MOL.19981007.0030.
Wilson, C.N. and Bruton, C.J. 1989. *Studies on Spent Fuel Dissolution Behavior Under Yucca Mountain Repository Conditions.* PNL-SA-16832 or UCRL-100223. Livermore, California: Lawrence Livermore National Laboratory. ACC: HQX.19891130.0045.

Efurd,D.W.; Runde, W.; Banar, J.C.; Janecky, D.R.; Kazuba, J.P.; Palmer, P.D.; Roensch, F.R. and Tait, C.D. 1998. Neptunium and Plutonium Solubilities in a Yucca Mountain Groundwater, Environ. Sci. Tech., 32, 3893-3900. Easton, Pennsylvania: American Chemical Society. TIC: 243857.

Wruck, D.A. and Palmer, C.D. 1997. *Analysis of Elevated Temperature Data for Thermodynamic Properties of Selected Radionuclides.* UCRL-ID-128955. Livermore, California: Lawrence Livermore National Laboratoty. ACC: MOL.19980109.0250..

Firsching, F.H. and Brune, S.N. 1991. "Solubility Products of the Trivalent Rare-Earth Phosphates". Jour. Chem. Eng. Data, 36, 93-95. Washington, DC: American Chemical Society. TIC: 240863.

Section 3.4 Criticality Evaluation of Configurations

3-10 Justify the use of the fresh fuel assumption in the internal criticality evaluation for waste forms other than commercial and naval SNF.

Many types of fuel that contain burnable poisons can be more reactive at moderate levels of burnup than when fresh. For such fuels, analysis of poison depletion and other burnup reactivity effects may be needed, not for burnup credit, but rather as a way of bounding the potential in-package burnup "debit."

The "fresh fuel assumption" from the Topical Report is intended to indicate that no reduction in reactivity from burnup will be accounted for in the analyses. The assumption also includes not taking credit for the presence of burnable absorber. DOE intends that the full justification of assumptions for waste forms other than commercial SNF be in the addenda to the Topical Report. This is noted in Section 4.1 of the Topical Report. A clear explaination of the "fresh fuel assumption" is planned to be provided in the addenda.

Preliminary information on the DOE-owned spent nuclear fuel waste forms has been presented to the NRC staff. From the Appendix 7 meeting at Idaho National Engineering and Environmental Laboratory (INEEL) on June 16, 1999 in the presentation titled "Disposal Criticality Analysis for DOE-Owned Spent Nuclear Fuel," the following conservatisms are planned to be utilized in the analysis of DOE-owned Spent Nuclear Fuel (SNF).

- Fresh fuel assumption (which includes a burnup penalty - no burnable absorber credit (the fuel does not contain burnable poisons))
- Maximum buildup of fissile isotopes
- Optimum moderation
- Optimum orientation
- Maximum absorber loss from geochemistry calculations using extremes of parameter ranges

In addition to the conservatisms listed above, the effects of isotopic decay are considered (Pu-239 to U-235, Pu-240 to U-236, etc.) to identify the most reactive isotopic composition.

For spent fuel other than breeder SNF, it is not possible for positive reactivity effects to result due to burnup if credit for burnable absorbers is not taken since the fissile inventory decreases. For breeder or plutonium-production SNF, the maximum gross buildup of fissile isotopes will be used to bound all burnup effects. Before SNF acceptance at the repository, verification that the fissile inventories fall below those evaluated will be required. These very conservative approaches will simplify the analyses by bounding the reactivity effects of burnup without requiring any extensive calculations or chemical assays. The general discussion in Section 4.1 of the Topical Report will be clarified to address this point. Detailed information is planned to be provided in the appropriate addendum to the Topical Report and/or individual validation reports.

Section 3.4.1 Computer Codes

3-11 Justify the use of the fresh fuel assumption in the external criticality evaluations for waste forms other than commercial and naval SNF.

Criticality evaluations for near-field and far-field configurations must consider the actual compositions of SNF materials. Using the fresh fuel composition would not be bounding for scenarios where uranium, plutonium, and other fissionable actinides have different potentials for mobilization and reconcentration.

DOE plans to justify the fresh fuel and other assumptions in the external criticality evaluations for waste forms other than commerial and naval SNF will be addenda and/or validation reports for the waste forms.

As discussed in the response to RAI 3-10, a conservative representation of the fissile content in the SNF is planned to be utilized. The burnup of DOE-owned SNF is low compared to commercial SNF, resulting in low production of transuranics. In addition, most of the inventory is medium-to high-enriched, also leading to low production of transuranics. As discussed in the response to RAI 3-10, the maximum gross buildup of fissile isotopics is planned to be considered for breeder or plutonium-production SNF.

Analyses to date (CRWMS M&O 1997 and CRWMS M&O 1998) have indicated that tens to hundreds of kilograms of uranium or plutonium are required to accumulate externally in order to create a critical configuration. The tens of grams of transuranics produced in the burnup of DOE-owned SNF are insignificant compared to the isotopics employed in the analyses.

CRWMS M&O 1997. *Criticality Analysis of Pu and U Accumulations in a Tuff Fracture Network.* A00000000-01717-0200-00050 Rev 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19980216.0260.

CRWMS M&O 1998. *Report on External Criticality of Plutonium Waste Forms in a Geologic Repository.* BBA000000-01717-5705-00018 Rev 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19980318.0412.

Section 3.4.2 Material Composition of Commercial SNF

3-12 Explain how the neutron-induced breeding of fissile and fissionable nuclides over time in the repository has been evaluated.

The TR does not indicate whether breeding effects have been evaluated. A scoping analyses of neutron sources, including (a,n) reactions, and associated breeding reactions should be provided or referenced in the TR. The evaluation should consider all fertile nuclides present in the various waste forms (e.g., U-238, Th-232, Pu-240). This RAI also applies to the material in section 3.4.3.

DOE agrees that consideration of the neutron-induced breeding of fissionable isotopes in the methodology is appropriate. The DOE plans to revise the Topical Report to note that the neutron-induced breeding of isotopes is to be considered. However, DOE believes that the evaluation results are application issues and therefore are not appropriate for inclusion in the Topical Report. The following discussion is provided for informational purposes only. The information provided is preliminary and may be updated as necessary with additional evaluations prior to the License Application.

Preliminary considerations on the effect of neutron-induced breeding of fissile and fissionable nuclides over time in the repository have been made. The neutron flux in a waste package is orders of magnitude smaller than that found in an operating reactor core at full power. Thus, the reaction rate required for conversion of fertile to fissionable isotopes is expected to be small in the waste package. During the long time periods that spent fuel would be in the potential repository, significant amounts of fissionable isotopes are not expected to be generated by this conversion. A complete evaluation of this assumption is planned to be provided as support documentation for the License Application.

Section 3.4.3 Principal Isotopes for Commercial SNF Burnup Credit

3-13 Explain why the verifiability of the inventory, as part of the isotopic validation, is not one of the criteria considered in selecting the principal isotopes for burnup credit.

As indicated in the report, nuclear, physical, and chemical properties of neutron-absorbing isotopes were considered in selecting them for burnup credit. The verifiability of isotopes for pre-closure configuration, in terms of isotopic validation, should also be one of the criteria in selecting the isotopes which can be used in subsequent post-closure isotopic inventory for criticality calculations.

DOE plans to address verifiability of isotopic inventory in the validation process. As is explained in the Topical Report, the principal isotopes selected for burnup credit are ranked by their expected reactivity contributions (both positive and negative) in discharged commercial SNF as a function of burnup. DOE plans to address isotopic validation in the commercial SNF validation reports. (The planned workscope for these reports is included as Attachment A.) This isotopic validation is expected to rely, in part, on comparison to radiochemical assay measurements. However, not all of the 29 principal isotopes are represented by the radiochemical assay data. For example, ^{233}U will not be present for many thousands of years but is included in the list to account for future positive reactivity addition. Some of the principal isotope fission products are also not represented in the radiochemical assay database. Thus, additional validation efforts are expected to include establishing the impact of potential isotopic concentration uncertainties on predictions of criticality. This process is part of the isotopic model validation requirements noted in Section 4.1.3.1.4 of the Topical Report. These validation efforts include the use of integral experiments (see responses to RAIs 4-4 and 4-5c) and are expected to address the impact of potential compensating effects in these experiments. This isotopic model validation process will be used to confirm the conservative aspects (bounding with respect to k_{eff}) of models developed for waste package design. These models are expected to based upon the required conservative input parameters for integral depletion and reactivity evaluations that minimize waste package criticality potential reguardless of waste form configuration.

With this bounding model approach and the plan for dealing with isotopic uncertainties, DOE believes that there is no need for a preclosure isotopic validation that would be subject to the verifiability of isotopic inventory. Therefore, DOE also believes a "verifiability of inventory" criterion is not necessary to be applied to the selection of principle isotopes for application to burnup credit.

3-14 Provide isotopic importance as a function of time which includes decay and loss of isotopes from spent fuel degradation in the "Principal Isotope Selection Report"

The degraded configuration described in CRWMS M&O 1998f indicates intrusion of water into the fuel rods without considering the reduction of isotopes through their dissolution in the water.

Isotopic importance as a function of time during time periods of spent fuel degradation where isotopes can be affected by the presence of water is not addressed in CRWMS M&O 1998f because this information is considered to be part of a degraded-mode configuration. Principal isotope burnup credit including the selected fission products is considered only for configurations in which the fuel is intact. With intact fuel there is no loss of isotopes through spent fuel degraded-mode mechanisms. For configurations in which the spent fuel is degraded, only actinides are considered. The presence and removal of actinides during degraded modes is being studied in detail. Isotopic importance as a function of time for actinides is expected to be quantified in these

degraded-mode studies rather than in the "Principle Isotope Selection Report." Additionally, degraded-mode studies are expected to be conducted to determine the uncertainties associated with the presence and removal of actinides at extended decay times.

3-15 Justify taking credit for the isotopes in Table 3-1 through verification of their quantities predicted by the isotopic model.

To assume that the spent fuel is composed of the 29 isotopes listed in Table 3-1, one must verify the quantity of the isotopes predicted by the isotopic models. The isotopic models predict the radionuclide inventory as the function of reactor operating history. This validation must be performed by direct comparison of calculated to the measured isotopic inventory. Under-prediction or over-prediction of an isotope will have a direct effect on predicting the criticality potentials of a waste package accurately.

DOE believes that the measured isotopic inventories (radiochemical assay information) and commercial reactor critical (CRC) integral experiment data results are not part of the methodology. However, DOE plans to usehese results , if the methodology is accepted, to perform the validation of the models in accordance with the methodology (See Appendix A for the planned workscope of the commercial SNF validation reports). Therefore, the following information is provided for informational purposes only.

As noted in the Topical Report (Section 4.1.1) and the response to RAI 3-13, radiochemical assay data will be used in verifying isotopic concentrations predicted by the isotopic model. DOE realizes that under-prediction or over-prediction of an isotope will have a direct effect on predicting the criticality potential of a waste package. DOE plans to perform comparisons of radiochemical assay data with calculated data to attempt toestablish variations in the isotopic concentrations predicted by the isotopic model. The sensitivity of variations in individual isotope concentrations in the model are planned be analyzed to establish the effect on criticality potential of a waste package. The methodology requirements for the isotopic model require that the isotopic model used for waste package design must produce isotopic concentration values that will ultimately result in conservative k_{eff} predictions for the waste package.

Additionally, CRC data will be used to further validate the isotopic model. The CRC data provides information on reactor operating histories that includes assemblies with strong neutron absorber history effects. This information will be used to supplement the radiochemical assay information that is often collected from "average" assemblies that do not have strong neutron absorber histories.

Thus, the accuracy of predicted isotopic inventory for the 29 isotopes listed in Topical Report Table 3-1 will be quantified by isotopic validation activities that include direct comparison of calculated-to-measured isotopic inventory, comparison to CRC data, and code-to-code comparisons. These isotopic validation activities are expected todividually justifytaking credit for the 29 listed principle isotopes (with the exception of U-233 and

Eu-151, which build in after long decay times). This credit is not sought in the Topical Report, but rather is expected to be sought in the License Application by reference to the validation reports.

Section 3.5 Estimating Probability of Critical Configurations

3-16 Answer the following questions related to computational feasibility and the discussion on page 3-21:

(a) How many repetitions or histories are envisioned for the Monte Carlo simulation?

(b) What is the confidence limit on a calculated criticality probability?

(c) How are the probabilities for different configuration classes combined?

(d) What will be done if one of the intermediate steps cannot materialize (i.e., a plausible regression form cannot be obtained)?

In order to establish a 10^{-4} probability, a very large number of simulation runs must be generated. Additionally, in calculating criticality probabilities, it is desirable to calculate the associated confidence limit(s) about that probability. Answers to these questions could provide a clearer picture of the described methodology for estimating the probability of critical configurations.

DOE believes that the questions on computational feasibility are application issues. However, in the interest of clarification, the following preliminary discussion is provided for informational purposes only.

a) DOE has used simulations with 50,000 to 1,000,000 repetitions. The latest versions of the Monte Carlo code can execute between 10,000 and 50,000 repititions per second (on a 300-MHz PC), so a run of several hours could produce 10^8 repititions, if the additional accuracy were required.

~~a)~~b) The probability of criticality will generally be expressed as the expected number of criticalities occurring before some specified time (typically 100,000 years). A confidence limit equal to 0.95 or 0.98 will generally be appropriate for such a parameter estimate. This confidence limit will correspond to a confidence interval of $\pm 1.98 \cdot \sigma$ or $\pm 2.33 \cdot \sigma$, respectively. The value of the standard deviation, σ, will reflect principal uncertainties associated with the Monte Carlo simulation: (1) the random fluctuations due to the limited number of samplings, (2) errors inherent in the regression or table lookup and interpolation process, and (3) uncertainty in the configuration parameters for processes that will take place over long time periods. For the first two uncertainty types the error can be driven as small as desired by increasing the number of repetitions or the number of points in the lookup table. The contribution of configuration parameter uncertainty to the overall standard deviation is determined by the probability distribution of such parameters. The standard deviation can be estimated by repeating the entire Monte Carlo runset with the random number seed randomized for each repeating, and then taking the standard deviation of the sample of repeatings. For the simulations run

thus far, with from 50,000 to 1,000,000 repetitions per simulation, the standard deviation computed in this manner is only a few percent of the mean (expected number of criticalities).

c) In the Monte Carlo process the probabilities are not really combined; rather, they are expressed by sampling from distributions that characterize the probabilies. If the probabilities are independent, the probability distributions will be functions of a single variable. If there are dependencies the probability distribution of one parameter will be a function of one or more other parameters. To properly apply such a conditional probability distribution, care must be taken to sample the independent parameter(s) before the dependent one(s). In this manner the value(s) of the independent parameter(s) can be used when sampling for the dependent parameter from the conditional probabiliity distribution.

d) If there are difficulties with intermediate mathematical steps, there are always alternatives. For example, DOE has already found difficulties with using the concept of a regression (or curve fit) for k_{eff} as a function of various degradation parameters. For waste forms with relatively high plutonium concentration, the sensitivities to neutron spectrum were confounding (reducing the determinacy of the regression coefficients) the degradation parameters actually being modeled (e.g. loss of fission products, loss of iron oxide). This problem was detected by our routine, careful comparison of results for different parameter sets. The alternative of table lookup and interpolation was used instead of the regression. For three or more parameters, and a large number of iterations, the table lookup and interpolation has increased the running time for each case. But it is still well under a minute for the scenarios internal to the waste package.

With respect to the question of the number of simulation runs (repititions) required for a probabilty of 10^{-4}, it should be noted that the *uncertainty* of a Monte Carlo estimate, expressed as a fraction of the parameter being estimated, will be approximately the reciprocal of the square root of the number of repititions. The 10^8 repititions that can be accomplished in a few hours (see (a) above) would be sufficient to drive the uncertainty to 10^{-4}.

3-17 Justify the assumption that Fe_2O_3 is the product that is formed by the corrosion of iron.

Credit is being taken for the filling of breached WPs by iron corrosion products, namely Fe_2O_3, thereby limiting the quantity of water present. It is unclear why the possibility that some of the iron corrosion product may be in the form of FeOOH was not considered. Justification of why the formation of FeOOH in lieu of Fe_2O_3 was not considered should be provided or else the effects of FeOOH formation on criticality control should be determined.

The methodology does not include an assumption of the presence of Fe_2O_3. All applications of the methodology have considered both Fe_2O_3 (hematite) and FeOOH (goethite). The body of the Topical Report makes no mention of either, using only the

generic name, iron oxide. The example given in Appendix C uses only iron oxide because our evaluations have shown they have approximately the same k_{eff} reducing effect for water moderated criticality. Both act primarily by displacing water. The hematite has higher density, and therefore does not displace as much water as an equal number of moles of goethite. This advantage of goethite is approximately compensated by the moderating effect of the hydrogen in goethite. In the EQ3/6 geochemistry analyses the hematite occurs when the goethite is suppressed and vice-versa. Although criticality evaluations have thus far not detected any significant neutronic difference between hematite and goethite (CRWMS M&O 1998, Tables 6.1-2 and 6.2-3), future evaluations are expected to continue to test for sensitivity between them.

CRWMS M&O 1998. *Criticality Evaluation of Degraded Internal Configurations for a 44 BWR Waste Pacakge*. BBA000000-01717-0210-00020 RE 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19989825.0207.

3-18 Indicate whether the criticality calculations will account for neutron interactions between WPs.

The calculation of the effective neutron multiplication factor, k_{eff}, should account for all fissile material that can impact the modeled system. The EDA-116 design places the WPs much closer together in the line-loading formation, which will lead to greater neutronic interaction between the packages. It is not clear from the topical report whether these effects will be accounted for when calculating the k_{eff} of the fuel both inside the WP and in the near-field.

The effects of interactions between adjacent waste packages and between waste packages and adjacent near-field accumulations will be documented in the supporting documents for the License Application. The worst case, of the lesser burned assembly ends facing each other, will be evaluated, using the axial dependencies described in the response to RAI 3-2, above. The validation of the isotopic model to support the axial dependence specification is described in Section 4.1.3.1.4 of the Topical Report.

Section 3.6.1 Type of Criticality Event

Slow versus fast reactivity insertion rate

3-19 Explain why the configuration with seismic event causing reshuffling of spent fuel and the spent fuel being fully submerged in the water inside the waste package is not considered as a plausible scenario for the fast reactivity insertion rate.

The reference cited in the topical report (CRWMS M&O 1997a) provides earthquake consequence analysis with respect to criticality in terms of iron oxide settling in the bottom of the waste package and providing the transient criticality analysis.

With respect to iron oxide, it has not been demonstrated that the iron oxide can remain in the waste package. Secondly, the reshuffling of the spent fuel assemblies during a seismic event is a more plausible scenario than the iron oxide mixing with water and becoming a homogenous solution. Thirdly, if even the iron oxide would remain in the waste package, the settled iron oxide configuration is the initial condition and the uniformly distributed configuration is the condition right after the seismic event. It is not clear how the scenario is postulated with these two conditions being reversed. Therefore, the reshuffling of spent fuel in the time frame of a second or less without the iron oxide is the more realistic scenario than the one presented in CRWMS M&O 1997a.

The evaluations referenced in the Topical Report are example evaluations to demonstrate how the methodology would work, not part of the methodology itself. The DOE believes that the specific types of configurations analyzed are application issues not appropriate for inclusion in the Topical Report. The following discussion is provided for informational purposes only. The information provided is preliminary and may be updated as necessary with additional evaluations prior to the License Application.

The re-shuffling of the fuel by a seismic event *was* considered a plausible scenario. However, it was not used as a model for the example transient criticality because preliminary calculations with intact assemblies indicated that there was no physically possible shuffling that could produce a sudden increase in k_{eff} from below to above the CL. It is still possible that a configuration with one, or a few, collapsed assemblies could result from such a shuffling. Such configurations are expected to be evaluated, and, if any is found to produce a significant increase in k_{eff} to above the CL, the transient consequences are expected to be evaluated at the higher insertion rate. It is expected that this behavior will only occur for a limited set of collapse patterns (those concentrated in the bottom of one stack of assembiles), so the probability of such a configuration is also expected to be evaluated for inclusion in the overall risk evaluation.

The shift from uniform to settled iron oxide was used as the nominal fast insertion scenario in the Topical Report example (Appendix C) because the latter configuration was significantly more reactive than the former. The question of particulate adherence to surfaces and entrainment in flow is so complex that DOE cannot state with high confidence which configuration would be more stable following a seismic upset. However, subsequent modification of the waste package design, i.e., adding aluminum thermal shunts, has altered the water displacement possibilities so that it appears there is now no longer much difference in reactivity between the two iron oxide distributions.

The issue of iron oxide solubility and whether it remains in, or flows out of a waste package is planned to be addressed in the degradation analyses and the detailed geochemical analysis. For the License Application, DOE expects to show that most of the iron oxide from steel corrosion remains in solid form. Evidence will be included from archeo-metallurgy, pictures and other records of examinations of sunken steel vessels, and experiments.

Steady-state versus transient

3-20 *Explain why, for transient criticalities, you cannot have conditions 1 and 2 met under a more realistic scenario with seismic event and partially-flooded waste package with no iron oxide*

The confining condition stated in the topical report is not needed for an impact of fast reactivity insertion in the waste form. With respect to the second condition, the fast reactivity insertion is plausible under the seismic event with the top row assemblies rolling over and being submerged inside the waste package or fissible materials reshuffling and coming together outside of the waste package. With regard to the third condition, even for the optimistic condition described in the topical report, the k_{eff} for inside the package is 1.0189 which is super prompt criticality. Therefore, the rate of energy release is very fast.

This RAI is apparently referring to the conditions stated in the third paragraph of Section 3.6.1 of the Topical Report, for a transient criticality having *significant kinetic energy release*. In this section the phrase *significant kinetic energy release* was intended to refer to a nuclear explosion. This paragraph was only intended to set the issue in perspective, and not to screen it out. The DOE plans to evaluate all recognized autocatalytic configurations for possibility of explosion, as discussed in the responses to RAI items 1-4, 3-21, 3-22, and 4-50, and in compliance with the direction in the letter from USNRC to Lake Barrett (Deputy Director OCRWM), August 7, 1995, *Review of Potential for Underground Autocatalytic Criticality*. It was not intended to indicate any diminution of the transient criticality effort using RELAP5, which is given as an example in Appendix C of the Topical Report. The Topical Report will be modified so that this intention is clarified.

With respect to the specific configuration suggestions of this RAI, DOE believes that comprehensive discussion of specific scenarios or configurations is an application issue rather than a methodology issue and therefore not appropriate for inclusion in the Topical Report. The following discussion is provided for informational purposes only. The information provided is preliminary and may be updated as necessary with additional evaluations or experiments prior to the License Application.

DOE has not evaluated the possible reactivity insertion rates for configurations having no iron oxide because it is physically impossible to lose all the iron oxide and still retain most of the fissile material in the waste package. This is because most of the iron oxide forms as a hard scale, not an easily poured or easily entrained fine powder (Angus et al. 1962, p. 956-968). As mentioned in the response to RAI 3-19, above, for the License Application DOE expects to show, in documents supporting the License Application, that most of the iron oxide from steel corrosion remains in solid form. Nevertheless, in the interest of conservatism DOE plans to evaluate possible upset conditions with a significant fraction of the iron oxide removed from the waste package.

Angus, N.S.; Brown, G.T.; and Cleere, H.F.; (1962) *The Iron Nails from theRoman Legionary Fortress at Inchtuthil, Perthshire,* J. Iron and Steel Inst.; November 1962, p. 956-968.

Under-moderated versus over-moderated

3-21 Justify why moderation was the only mechanism used to govern the positive or negative feedback characteristics of a critical system.

The topical report's current discussion does not recognize, for example, that in certain configurations water is a poison and that other moderators (SiO_2) more strongly influence the thermal neutron spectrum. Furthermore, particle self-shielding mechanisms for absorbers and fissile materials can have important implications not normally associated with the concept of over/under-moderation. Reflection dynamics likewise may be important in certain scenarios.

The concept of under/over-moderation has limited applicability outside LWR cores. For example, the 1986 Chernobyl disaster, by far the worst autocatalytic criticality event in history, was governed by positive void reactivity effects that have nothing to do with the concept of over-moderation. The positive void reactivity effects in CANDU reactors are likewise unrelated to over-moderation.

Especially in configurations where positive feedback effects are deemed credible, it is important to analyze the dynamic progression of criticality events using appropriately coupled models of the actual neutronic and thermal-mechanical phenomena in that system. Repository physics can differ fundamentally from LWR core physics. Correct analysis of the criticality dynamics is essential to assessing any potentially disruptive effects in the repository.

The methodology determines mechanistically the possible configurations. The feedback mechanisms or conditions associated with each configuration will be accounted for. The examples evaluated in the Topical Report are not intended to be all-inclusive. Their purpose was to demonstrate how the methodology could be applied, not all the conditions to which it would be applied. DOE believes that comprehensive discussions of specific scenarios or configurations is an application issue rather than a methodology issue, so they are not appropriate for inclusion in the Topical Report. The following discussion is provided for informational purposes only. The information provided is preliminary and may be updated as necessary with additional evaluations or experiments prior to the License Application.

With respect to moderators other than water, the potential for silica moderation has already been considered in preliminary analyses. The internal criticality potential is largest in the codisposal waste package because of the large amount of glass. The codisposal criticality evaluations have included have always included the moderating effects of any silica present, and have been carefully analyzed for possibible autocatalytic effects. None has been found thus far, but this screening is expected to continue. For

external accumulations of fissionable material, our analyses thus far have found that although the amount of silica in the rock may be large, the moderating effect is still small compared to water. (CRWMS M&O 1998)
The transient criticality example in Appendix C of the Topical Report uses RELAP5, which considers the void effects suggested in this RAI. The only type of criticality consequence calculation that considers only moderator feedback is the steady-state criticality, for which it is the dominant feedback mechanism, as is explained in Section 5.1 of Appendix C of the Topical Report. For transient criticality external to the waste package, the Topical Report states that a code with fully coupled thermal, hydraulic and neutronic effects will be used (Section 4.4.1.2). The DOE plans to specify the code to be used, when it is determined, in the validation report for the consequence model.
CRWMS M&O 1998. *Report on External Criticality of Plutonium Waste Forms in a Geologic Repository.* BBA000000-01717-5705-00017 Rev 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL 19980318.0412

Section 3.6.2 Evaluating Direct Criticality Event Consequences

3-22 Justify the statement that accumulation and geometry of fissionable mass needed for large disruptive criticality events is expected to be beyond anything physically possible in the repository.

It is not clear why the statement is made when another statement in the same paragraph explains that "some theoretical analyses have identified larger, disruptive consequences ..."

The statement alluded to was intended to convey the fact that all of our analyses thus far (CRWMS M&O 1997, Section 9.1) fail to support the accumulation of the autocatalytic configurations suggested in the literature (either by Bowman or the group at the University of California Berkley Nuclear Engineering Department, cited in Section 4.4.1.2 of the Topical Report). The statement was for informational purposes only, and is not important for justifying the methodology. Specific reference to the analyses will be provided in the revision to the Topical Report, which will be clarified to remove the apparent contradiction.

CRWMS M&O 1997. *Waste Package Probabilistic Criticality Analysis: Summary Report of Evaluations in 1997.* BBA000000-01717-5705-00015 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19980204.0095.

Section 3.7 Estimating Criticality Risk

3-23 Justify the assumption that the only detrimental effect of a criticality event on the repository performance is the generation of additional radionuclide inventory.

In addition to the increase in radionuclide inventory, other direct and/or indirect

potential criticality consequences must be considered. Increase in the waste package heat output affects the near-field environment and the rate of material corrosion and waste form degradation within the waste package. Additionally, large disruptive criticality transients could generate sufficient heat and pressure to degrade the waste package, cladding, or spent fuel. This degradation of the waste form could increase the release rate of radionuclides and the corresponding dose at the critical group location. This comment also applies to Section 12, Section 4.4.1.2, Section 4.4.1.1, Section 4.4.1.2, and Section 4.5.

Any statement about increased radionuclide inventory was intended to apply to the steady-state criticality. DOE will revise the Topical Report as necessary to correct any unintentional implication that additional inventory is the only effect of criticality determined. The Topical Report devotes considerable attention (Section 4.4.1.2) to the transient criticality consequences of temperature and pressure, which could cause changes in the near-field environment, changes in the corrosion rates of the waste package materials, and damage to the waste form as suggested in this RAI. The peak transient overpressure calculation mentioned in the Topical Report (Section C-5.1) is not compared to a specific criterion for this damage. However, future calculations of these parameters are expected to include such a comparison.

Chapter 4.0 Model Description
Section 4.1.1.2 Postclosure Isotopic Concentrations

4-1 Explain how the so-called bounding bias and uncertainty values are derived from a stochastic process.

For example, are statistical confidence intervals associated with the uncertainty values? Are these a function of the number of Monte Carlo histories? In which way is the Monte Carlo analysis used to derive these values of bias and uncertainties?

At least 1000 trials are run, resulting in a normal distribution of reactivities which reflect the effects of the decay and branching ratio uncertainties. The exact number will depend on when convergence is shown. Therefore, the uncertainty values do have statistical confidence intervals associated with them that are a function of the number of Monte Carlo histories. The mean and standard deviation are calculated for these trials. The bias, standard deviation, and number of trials run are used to determine a one-sided tolerance limit which characterizes the population. Monte Carlo analysis uses the standard statistical methods for one-sided tolerance limits to calculate the bias and uncertainties.

An explanation will be added to a revision of the Topical Report to clarify the points made in this RAI.

Section 4.1.3 Neutronic Model Validation

4-2 *Justify the applicability of Commercial Reactor Criticals (CRC) for validation of MCNP4B in light of the lack of cross section libraries as a function of temperature.*

It is not clear how well the MCNP4B cross sections can be validated against CRC when the modeling of CRC requires codes with cross sections as a function of temperature.

Although the MCNP cross section library is limited in available temperatures, it does offer a variety of temperatures for many isotopes. Included in the list of isotopes represented at different temperatures are both U-235 and U-238. Both of these isotopes are modeled in the CRCs using the ".53c" suffix. This represents the cross sections at a temperature of 587 K. Also, the moderator cross sections are corrected for temperature using the scattering-law treatment (the $S_{\alpha,\beta}$ card) provided in MCNP. The cross sections of the structural material are not significantly affected by temperature. The remaining isotopes are either not signficantly affected by temperature, or are in such small quantities that the temperature affects on k_{eff} related to these isotopes are not signficant.

The effect of temperature limitations in MCNP will be analyzed as part of the waste form-specific validation reports. The Critical Limit development is expected to consider the effects of temperature-related uncertainties. Quantification of any temperature-related bias is expected to include a code-to-code comparison between MCNP and KENO, which has cross sections that can easily be processed for temperature. (See Attachment A, Part II.C.)

Section 4.1.3.1.3 Radiochemical Assays

4-3 *Provide information on the initial enrichments and burnup for the new Radiochemical assay measurements that are being conducted to supplement the database for commercial SNF isotopic model validation.*

Staff notes that the existing data are limited to enrichments between 2.45 and 3.87 wt% ^{235}U. The new data should be for higher initial enrichments and higher burnup.

DOE believes that the radiochemical assay information is not part of the methodology. It is information that will be used, if the methodology is accepted, to perform the validation of the models in accordance with the methodology (See Appendix A for the planned workscope of the Commercial SNF Validation Report). The following discussion is provided for informational purposes only.

Additional isotopic assays are being performed for both PWR and BWR fuel rod samples. The PWR fuel rod samples were obtained from the TMI-1 reactor at the end of cycle 10. The BWR fuel rod samples were obtained from Quad Cities 1 reactor at the

end of cycle 12. Both Argonne National Laboratory (ANL) and General Electric Vallecitos (GE) are performing radiochemical assay measurements on the fuel rod samples. The table below summarizes the enrichments and burnups for the fuel rods that were analyzed. DOE believes that with these additional assays, sufficient assay data will be available to support validation of the analysis method.

Reactor Type	Fuel Rod ID	Rod Average Burnup* GWd/mtU	Fuel Rod Enrichment (wt% U-235)	Number of Samples Analyzed ANL/GE
PWR	O1	28.5	4.65	0/3
PWR	O12	28.0	4.65	0/3
PWR	O13	27.0	4.65	0/2
PWR	H6	48.5	4.00	5/0
BWR	A2	~70.0	3.80	0/2
BWR	B1	~70.0	3.80	2/2
BWR	C7	~70.4	3.00	3/1
BWR	G5	~62.0	3.00/2.00**	2/1

* Peak rod burnup is approximately 10% greater.
** Rod contained 2.00 wt% gadolinium oxide mixed with fuel.

Section 4.1.3.1.4 Requirements for Isotopic Model Validation

4-4 Justify the use of 45 reactor core state points to bound the spent fuel operating history parameter values of the historical and projected spent fuel discharge for the spent fuel assemblies, which are destined for disposal in the proposed repository.

The operating history parameter values of the 45 reactor core state points do not bound the operating history parameter values of the 100,000 or so commercial spent fuel assemblies, which will be placed in the proposed repository at Yucca Mountain. The bounding operating history parameter values must be established based on the operating history parameter values of the historical and projected spent fuel assemblies to be discharged from the reactors.

DOE believes that the data associated with the 45 reactor core criticals are not part of the methodology. Rather, these data will be used, if the methodology is accepted, to perform the validation of the models in accordance with the methodology. (See Appendix A for the planned workscope of the Commercial SNF Validation Report.) The following discussion is provided for informational purposes only.

Development of the bounding model (bounding operating history parameter values) to be used in models for the prediction of waste package criticality potential will draw from many sources of data (e.g., radiochemical assay data, CRCs, utility/vendor databases, numerical experiments) describing spent fuel assemblies and will not be limited to 45 reactor core criticals. The bounding model will not rely upon core average values drawn

from reactor core criticals. Once a bounding model is defined it will be validated in the Commercial SNF Validation Report.

The 45 CRCs will not be the sole source of information used to bound spent fuel operating history values for historical or projected spent fuel discharge. These CRCs will be used within their ranges of applicability to develop bias and uncertainties for the depletion and reactivity models (integral experiment) that will be used in waste package design. The range of applicability for this calculated bias and uncertainty will be extended to an appropriate cross section of the existing and projected spent fuel assemblies which will be placed in the proposed repository. For spent fuel assemblies that fall outside of this range of applicability, alternate modeling methods and classifications will be used to ensure conservative prediction of criticality potential (fresh fuel assumption, etc.).

The 45 reactor core statepoints addressed in the Topical Report cover an initial fuel enrichment range of 1.93 – 4.17 weight-percent (wt%) of U-235, a core-average fuel burnup range of 0 – 33 gigawatt-day per metric ton of uranium (GWd/mtU), and an assembly-average fuel burnup range of 0 – 49 GWd/mtU. These statepoints represent 6489 fuel assemblies, of which 1,060 are unique burned assemblies. The statepoints were specifically chosen with fuel assemblies that are representative of a large percentage of the existing PWR commercial spent fuel assembly inventory. For example, the PWR CRC database represents several fuel assembly designs and was chosen to include fuel assemblies with control rod histories, axial power shaping rod histories, and burnable absorber rod histories.

Additionally, the reactivity effect of past in-reactor irradiation history for individual fuel assemblies (selected from the PWR database) is expected to be quantified with the numerical experiments. For example, an individual fuel assembly that is being evaluated is expected to be modeled assuming two or more neutron absorber histories for in-reactor irradiation. The numerical experiments are expected to be used for comparing small, localized perturbations of these models to existing measurements. The effect on k_{eff} is expected to then be quantified and documented in the validation report (see Attachment A). In addition, the results from these analyses are expected to be used, in part, to define the bounding model (i.e., bounding operating history parameters) to be used for waste-package design.

Radiochemical assay data will also be used in addition to the CRC statepoint data in defining bounding operating history parameters for use in calculating the commercial SNF isotopic concentrations used in criticality evaluations for waste-package design. DOE expects to demonstrate that the defined operating history parameters predict isotopic concentrations that (when used in criticality evaluations) bound (with respect to k_{eff}) similar criticality evaluations using either measured radiochemical assay data or best-estimate isotopic concentrations.

The Topical Report is seeking acceptance that the requirements presented in Section 4.1.3.1.4 are applicable for isotopic model validation for commercial SNF. These

requirements are part of the methodology. The validation and acceptability of the isotopic model is beyond the scope of the Topical Report and is expected to be demonstrated in the validation reports.

4-5 Justify the method used to determine the isotopic code bias.

The purpose of code validation is to quantify the bias and the uncertainty, which may exist within the isotopic code. The main problems with the approach described in 4.1.3.1.4 are:

a) Not using ANSI/ANS 8.1 and 8.17 to establish area and range of applicability.

b) Using CRC operating history, which is insufficient to cover the complete range of operating history parameters of the discharged PWR spent fuel assemblies destined for disposal, to establish the bounding parameter values.

c) Using the integral k_{eff} approach, which takes advantage of compensating errors in isotopic prediction to validate the isotopic model.

d) Using the established parameter values from Part a to perform calculation-to-calculation comparison, as opposed to comparing calculations to experimental results for the purpose of isotopic model validation.

Other approaches such as direct comparison of measured to calculate values would eliminate some of these concerns.

The following discussion will provide justification for the method used to determine the isotopic code bias. The Topical Report will be revised as appropriate to clarify the concerns raised by the RAI.

(a) DOE believes its method for establishing area and range of applicability is consistent with ANSI/ANS 8.1 and 8.17. Section 4.3.6 of ANSI/ANS-8.1-1998 will be followed for the validation reports for PWR and BWR SNF. Section 4.10 of ANSI/ANS-8.17-1984 will also be followed for the validation reports. This section states:

> "Credit may be taken for fuel burnup by establishing a maximum fuel unit reactivity and assuring that each fuel unit has a reactivity no greater than the maximum established reactivity by (1) a reactivity measurement or (2) analysis and verification of the exposure history of each fuel unit. Consideration shall be given to the axial distribution of burnup in the fuel unit."

The range of applicability of bias and uncertainties calculated using CRC integral experiments is expected to be determined during validation activities and documented in the validation reports. The fuel assembly burnups (exposure) for the CRC analyses are

taken from Core Operation Reports and reflect the exposure history of the fuel assembly. The exposure history is reflected with the presence or absence of neutron absorber material (e.g., soluble boron, burnable poison rods, and control rods), moderator density, and fuel temperature. The exposure history is also reflected in the axial distribution of the burnup. These unique, burned assemblies gathered from Core Operation Reports are expected to be used (along with radiochemical assay data, numerical experiments, and code-to-code comparisons) to define and confirm a waste-package design model that will yield a maximum fuel assembly reactivity (i.e., to satisfy Requirements A and B in Section 4.1.3.1.4 of the Topical Report). Also see the response to RAI 4-4.

(b) Bounding operating history parameter values for PWR (and BWR) spent fuel assemblies are expected to be developed using CRCs, radiochemical assay comparisons to calculated isotopic values, code-to-code comparisons, and numerical experiments. The development of this bounding model will not be limited to CRC data only. A range of applicability is expected tobe determined for the bounding model,and PWR spent fuel assemblies that are found to be outside of that range of applicability are expected to be treated with alternative conservative model parameters (fresh fuel assumption, etc.). See response to RAI 4-4.

(c) The DOE recognizes that using the integral k_{eff} approach requires investigation of potential compensating errors in isotopic prediction. For all cases, both CRC and radiochemical assays, the isotopic concentrations are expected to be compared for measured data (for radiochemical assays), best-estimate calculations, and application model calculations to relate the observed effect on k_{eff} to the change in isotopic concentrations. These comparisons are expected to quantify both individual and integral istopic effects on k_{eff}.

(d) Calculation-to-calculation comparisons are expected to be made during isotopic model validation. However, for all cases where the calculation-to-calculation comparisons are made, measured data (either k_{eff} or isotopic concentrations) also exists and is planned to be used.

4-6 Explain why k_{eff} adjustment approach, which takes advantage of compensating errors in isotopic inventory, is chosen over the direct adjustment of each isotopic inventory for capturing the isotopic decay and branching ratio uncertainties.

The most important parameter governing criticality potential in the waste package will be k_{eff}. Waste package design and waste package loading at the proposed repository will focus on this parameter. Because of the primary importance of this criticality parameter, the k_{eff} adjustment approach has been chosen. The effects of compensating errors will be quantified in the validation report for commercial SNF fuel assemblies (see Attachment A). Quantification of compensating errors will include effects on isotopic inventory as predicted by the depletion code and the corresponding effect on k_{eff} predicted by the integral calculation (CRC).

While some part of the time the effect of one error may "compensate" for the effect of another error, these errors are behaving within statistical bounds. Other licensing applications, such as core design using statistical methods and the quantification of measurement system errors, have utilized this accepted technique. However, applying direct adjustments to each isotope may not reflect the fact that the inventory of an individual isotope can be dependent on the inventory of other isotopes.

A method could be chosen to adjust the isotopic inventory of each isotope in the conservative direction (increase for fissile and decrease for absorber) by an amount that might potentially bound the reactivity contribution for all fuel configurations and burnups. However, this would require examining the uncertainties in the precursors of each isotope (not limited to the principle isotope set), as well as the uncertainty in the decay of that isotope for all fuel configurations and burnups and could be a source of additional error in calculations of isotopic inventory. The application of the resulting adjustment factors would require separately modifying all calculated isotopic concentrations before they are used in any criticality calculation, including configurations external to the waste package.

Radiochemical assays would be one source of data used to develop these adjustment factors. Errors inherent to radiochemical assay measurements would be introduced into the calculation of criticality potential based on the adjusted isotopic inventories. Additionally, isotopic concentrations determined from radiochemical assay measurements rely upon "micro scale" data (concentrations determined from a small portion of a fuel pellet) and are most often not accompanied by detailed depletion histories for the fuel assembly being sampled. Reliance upon this type of data for developing isotopic inventory adjustment factors would be risky.

Modification of isotopic concentrations during burnup requires a detailed knowledge of isotopic depletion interdependence. Likewise, modification of isotopic concentrations at discharge burnup requires a detailed knowledge of isotopic decay interdependence. The development of adjustment factors to be applied to isotopic inventory at discharge burnup would require capturing all of the isotopic decay interdependence detail as well as the uncertainties associated with isotopic decay and branching ratios. Development of these factors could prove to be a formidable task because of the complicated decay schemes common to exposed fuel isotopes and would more than likely introduce additional error into the calculation of isotopic inventory to be used in criticality evaluations. A very small error in isotopic inventory introduced at discharge burnups could translate to a very large error in predicted isotopic inventory after thousands of years of decay. Additionally, these adjustment factors would be difficult to relate to physical examples for verification purposes. Therefore, to minimize the introduction of additional error, and to focus on the most significant characterizing parameter of waste package criticality potential, DOE chose the k_{eff} adjustment approach.

As part of the planned validation report effort described in Attachment A, an approach has been chosen to quantify the effects of decay uncertainties, branching ratio uncertainties, and isotopic distribution variations inherent to the depletion model code.

This approach will quantify effects of compensating errors in the integral experiment on k_{eff} calculations. These compensating effects are expected to be well understood and characterized. See response to RAI 4-5 (c).

Section 4.1.3.2 Determination of Critical Limits

4-7 Provide justification for not incorporating the following information into Figure 4-1 for estimating Critical Limit.

(a) Identification of subsets of validation experiments which are applicable to the waste form and configuration classes within and outside the waste package.

(b) Performance of a normality test prior to applying any of the statistical analyses such as regression analysis, which is based on the normality assumption. Figure 4-1 shows that the normality test is performed after the regression analysis indicates there is no trends. The base assumption for regression analysis is normality which must be verified through some statistical tests.

(c) Performance of a regression fit of k_{eff} on predictor variables for the relevant subset to identify trending parameters.

(d) Inclusion of all the parameters, not just the ones with "strongest correlation," which have statistically significant trends as the function of k_{eff}.

(a) The first box on the top of Figure 4-1 incorporates this idea ("Define set of validation experiments … encompassing desired range of applicability"). The waste form-specific validation reports are expected to document the waste form and configuration-class specific benchmark subsets. These will be developed from the total benchmark database for each applicable scenario/waste class from the master scenario list. The subset development will consider such aspects as material type, geometry and neutron spectrum. (See Attachment A, Part II.A.)

(b) The first step in the assessment of a critical limit is to look for a trend of k_{eff} on a parameter. This involves a regression analysis.

Several assumptions form the basis for inferences regarding regression analysis, including the significance of the regression. Normality of the regression residuals is one assumption. It is not possible to assess normality of the residuals before performing the regression fit, by definition of residual. Most statistical software packages include tests of normality of the residuals as standard or optional output for regression calculations.

For many statistical inferences using the regression model, results are reasonably robust with respect to normality. Another assumption is the equality of the variances of these normal distributions at each value of the independent variable. This assumption can be checked with various residual plots. This is a more stringent assumption than the normality assumption for the prediction/inference aspects of regression. Another

assumption is that the independent or predictor variable is known without error, or with error that is very small with respect to the variability of the independent variable.

Figure 4-1 indicates that if the regression is not significant, then the recourse is to use statistical tolerance limits based on the k_{eff} values for the set of critical experiments. There is no predictor variable, because the regression is not meaningful, and the critical limit is developed without accounting for any trend of bias. For this situation, the normality test indicated in Figure 4-1 is for these k_{eff} values (not the residuals) since there is no regression, and the result determines the method for determining the critical limit that is appropriate. If the normality test of the k_{eff} values does not reject the hypothesis of normality of the k_{eff} values, then the Normal Distribution Tolerance Limit (NDTL) method is appropriate. If the hypothesis of normality is rejected, the distribution-free method (DFTL) will be used.

(c) The third box down from the top of Figure 4-1 incorporates this idea ("Perform regression fits of k_{eff} on predictor variables to identify the trending parameter"). The trend analyses performed for the waste form-specific validation reports will consider the benchmark data in appropriate waste form/waste class-specific subsets of the total benchmark database. (See Attachment A, Part II.B.)

(d) The approach chosen for determination of critical limits assumes the bias of the criticality computer code calculations can be adequately represented by trending on one predictor variable at a time. If the regression is not significant for any of the candidate predictor variables, one of the other tolerance limit techniques (NDTL or DFTL) will be applied.

The limited data used for providing examples of computing critical limits in the Topical Report did not include instances in which the regression could be considered significantly stronger by the addition of one or more predictor variables.

Extending the number of predictor variables would require describing additional techniques to provide the critical limit(s) with the desired statistical properties. For nonlinear models a meaningful tolerance band can not be provided, and, there are difficulties in assessing the usual statistical characteristics that are well known for linear regression models.

This added complexity of trending other than simple linear regression introduces other difficulties in the definition of trend that seem unwarranted in the limited data evaluated to date. With multiple predictor variables, the problem of collinearity (highly correlated predictor variables) must be considered. For a given set of criticality experiments, the range of the computed k_{eff} is generally going to be very small.

The application of the critical limit will likely be based on limited neutronics knowledge about the waste form that is to be evaluated. In many cases it is likely that there will not be sufficient waste form information to support use of multiple predictor varibles. Experience in the calculation of critical limits for sets of critical experiments has not

provided any encouragement to go beyond one predictor variable. DOE believes it is reasonable, practical, and sufficient to use a single predictor variable that is expected to be available for a given waste type.

It is a good practice to consider the direction of the trend. If there are two potential predictor variables, one with a positive correlation, and the other negative, then consideration should be given to conservatism. A negative trend would indicate that the criticality code is underestimating k_{eff}. If the "negative trend" predictor has the lower correlation with k_{eff}, it would also have a larger standard error of regression, resulting in added margin in the critical limit. While the possibility of two potential predictor variables is not explicitly addressed in the Topical Report, DOE plans to include this subject in the validation reports.

4-8 Provide the technical bases (other than "commonly used") for using 0.05, instead of 0.01 or 0.001, for the level of significance in identifying linear trends with respect to the trending parameters.

Although it is indicated that approval of a specific value for the level of statistical significance will be sought in the License Application, the TR should provide a statistical rationale used for selecting the specific value.

The level of significance is the probability of the error made when one rejects the null hypotheses that the slope of the regression fit is zero, given that the slope is zero. Setting this value to less than 0.05 provides for acceptance of the null hypothesis (lack of trend) more often than if a lower level of significance is chosen. The choice is made on the level of risk that is appropriate for the rejection of the null hypothesis that there is no significant slope, which is equivalent to no trend. For the trending of k_{eff}, the level of risk chosen is 0.05, which is a practical level of risk for rejection of trending. The rejection of trending results in the use of the Normal Distribution Tolerance Limit or the Distribution Free Tolerance method.

4-9 Justify the basis for redefining Δk_m.

ANSI/ANS-8.17 defines Δk_m as "an arbitrary margin to ensure the subcriticality k_s." The examples provided for Δk_m in the topical report such as "1) the effect on k_{eff} associated with the long-term decay of radionuclides in the waste form and 2) the effect on k_{eff} associated with extending the range of applicability of the CL beyond the experimental database" are standard biases, which must be included as part of isotopic bias and $\Delta k_c(x)$, respectively. The Δk_m in this case must include a subcritical margin. For example, if CL for a particular configuration is established to 0.95, is the value of 0.9499 for $k_s + \Delta k_s$ considered to be subcritical? If more neutron histories are used, the calculated value could be 0.95 or beyond. Therefore, the need to identify a zone of criticality and incorporate it into the total uncertainty should be considered.

This comment also applies to Normal Distribution Tolerance Limits (Section

4.1.3.2.2) and the Distribution Free Tolerance Limit (Section 4.1.3.2.3).

DOE agrees that Δk_m was not used consistent with ANSI/ANS-8.17. The Topical Report will be revised to clarify the differences and the reasons for them. However, the risk-informed methodology for disposal is not trying to ensure subcriticality as did the past applications of ANSI/ANS-8.17. The risk-informed methodology is defining a Critical Limit (CL) which establishes what is and is not critical. The past applications of ANSI/ANS-8.17 defined an upper subcritical limit that ensured an arbitrary subcritical margin. The CL values will not include an arbitrary subcritical margin (i.e., Δk_m as defined in ANSI/ANS-8.17). The disposal methodology is performing a screening process by identifying what general configuration classes have or do not have the potential for criticality. Beyond this screening, an extensive investigation is made to identify configurations with the potential for criticality.

With respect to Sections 4.1.3.2.2 and 4.2.3.2.3, the establishment of statistical tolerance limits is based on a single random sample from a population. For the same reasons as discussed in the previous paragraph, an arbitrary subcritical margin is not an appropriate part of a risk-informed methodology for postclosure criticality analysis.

4-10 Justify the use of the linear regression model to fit the data presented in Figure 4-2.

Considering the data in Figure 4-2, it seems that another model, i.e., exponential or polynomial, could better fit the data than the proposed linear regression model for trending criticality level.

The data in Figure 4-2 are illustrative, and the justification of the appropriateness of a specific model for establishing a critical limit for a range of applicability is expected tobe provided in supporting documents for the License Application. The following discussion is preliminary and is provided for informational purposes only.

The methods used to justify a linear regression trending may include comparison to competing models, but there is also a practical limitation to the complexity of trending a result such as k_{eff}. The choice of the trending model includes consideration of the concern for quantifying the behavior of the criticality code over a range of applicability for a waste type. If the figure of merit for for this is a statistical tolerance limit for the regression, only linear regression can be used. Linear models include polynomials and multiple predictor variables. But the preference, as stated in the Topical Report, is the simple linear regression, for ease of understanding of the application.

DOE's experience with regression fits of data similar to those in Figure 4-2 for a polynomial of degree two generally indicates no practical increase in the coefficient of determination (R^2) or decrease in the standard error. The exponential model (of the form $y = a * \exp(bx)$ can be made a simple linear regression by taking logarithms. The resulting regression does not improve the fit R^2, and it further complicates the use of the

results, since the results have to done in logarithms and then transformed back into the original unit.

Trending on the k_{eff} values for neutronics parameters on the basis of simple linear regression is the method illustrated in the Topical Report. The concept of trending is not the discovery of a complex relation of the calculated and observed, but to address the practical issues regarding the ability of the criticality code to match experimental determinations for k_{eff}. Therefore, DOE believes that a simple linear model adequately describes system behavior.

As noted in the response to RA 4-7, it is a good practice to consider the direction of the trend. If there are two potential predictor variables, one with a positive correlation, and the other negative, then consideration should be given to conservatism. A negative trend would indicate that the criticality code is underestimating k_{eff}. If the "negative trend" predictor has the lower correlation with k_{eff}, it would also have a larger standard error of regression, resulting in added margin in the critical limit. While this subject is not explicitly addressed in the Topical Report, it is expected to be included in the validation reports.

4-11 Justify why a single predictor is used for the least-square fits, as explained in the discussions. Examine the data to determine whether a combination of factors would yield a better fit.

One could argue that a "less sensitive" model (a model that does not include all significant factors and factor combinations, or a model with a nonlinear structure) is more conservative. This argument would be correct with respect to the measure of uncertainty, since a poor fit is associated with larger uncertainty. However, a more refined regression could have a negative trend that may be undetected due to the simplicity of the model. Therefore, the question is whether the failure to detect a negative trend is outweighed by the large measure of uncertainty.

A single predictor is used for the least-squares trending because if the trend is negative and significant, the results will likely display this result. If such a negative trend is not statistically different from zero, failure to detect it should have no practical impact on the resulting critical limit that would be determined using NDTL or DFTL methods. If there is a large measure of uncertainty, then it is unlikely that a more complicated model is going to be more useful than one of these two latter approaches. This position is based on the fact that large uncertainties will result in rejection of trending. Note also the response to RAI 4-7, above.

4-12 Explain how parameters other than those used for trending are applied to characterize a system and the benchmark experiments.

The extension of the range of applicability (ROA) must be addressed with caution. How would one know of any trending effects outside the experimental ROA?

Parameters, other than those used for trending, will be applied in tabular form to characterize a system and the related benchmark experiments in tabular form. For each parameter, the anticipated range of conditions and the benchmarked range of conditions will be documented, in a single table, for easy comparison. The non-trended parameters may include, but are not limited to: cooling time, fuel temperature, fuel type, cladding type, fuel density, geometry type (e.g., square lattice, triangular lattice, homogeneous solution), absorber types, and absorber concentrations.

The method for extending the range of applicability is dependent on the parameter and the specifics of any trend. Procedures for extending the range of applicability are expected to be developed, and will be specific to waste form, configuration class, and parameter. Where a trend exists, these procedures will consider the trend itself, the direction of the trend, and the physical explanation of the trend. Short of adding additional benchmark experiments, the extension process may include a code-to-code comparison in addition to physical explanations for the trend and statistical analyses of the trend. The procedure is also expected to include a method for determining the penalty ($\Delta k_{penalty}$) to be included in the extended range. These procedures are expected to be developed based on the guidance of ANSI/ANS 8.1, Appendix C and are expected to be documented in the validation reports. (Also see the response to RAI 4-22.)

Section 4.1.3.2.1 Lower Uniform Tolerance Band

4-13 Justify why the application of combined Method 1 and 2 in Lichtenwalter et al (1997, pp 158-162) as referenced in the topical report was not evaluated.

As stated in Lichtenwalter et al., "the recommended purpose of method 2," Lower Unifirm Tolerance Band (LUTB), "is to apply it in tandem with Method 1..." The term Δk_m must be included in the LUTB approach. However, the value for Δk_m may be determined based on some reasoning as opposed to the traditional 5% administrative margin.

The DOE believes that Δk_m is not relevant to the postclosure criticality analysis methodology for the reasons addressed in the response to RAI 4-9.

The application of the combined Method 1 and 2 in Lichtenwalter et al. does not reflect the repository application to populations of waste material. Lichtenwalter et al. was intended for transportation packages, whereas the topical report addresses a geological repository. These applications differ in design and regulation. The repository will hold populations of multiple waste forms, each of which is subject to a critical limit. Tolerance limits provide a specified confidence and proportion of the population that are of interest. The method chosen allows, for either trending or non-trending cases, providing a critical limit that has a stated confidence that a stated proportion of the calculated k_{eff} values will be below that limit.

Section 4.1.3.2.2 Normal Distribution Tolerance Limits

4-14 Clarify if the Normal Distribution Tolerance Limits (NDTL) are based on the prediction interval or tolerance interval, and justify this approach.

The prediction interval is based on predicting, with a predetermined confidence level, a single future value which would be below the critical limit. On the other hand, tolerance limits predicts a percentage of future values which would fall below the critical limit. The latter is a more acceptable approach.

The NDTL is a statistical tolerance limit for the set of k_{eff} values that represent a waste form and is not based on prediction of trending.

The NDTL method assesses the capability of the criticality code to predict k_{eff} values as a single figure of merit encompassing all the evaluations for the set of criticality experiments in which there is no identified trending parameter.

To determine the NDTL, the k_{eff} values for the critical experiments representing that waste type are tested for normality. If normality is found to be reasonable, the usual method for establishing a statistical tolerance one-sided tolerance limit is applied to determine the critical limit. This calculation is of the general form

$$CL = k_{ave} - k(\text{Confidence, Coverage, sample size})\ s_{combined}$$

where k_{ave} is the average value of the k_{eff} for the set of critical experiments, k(Confidence, Coverage, sample size) is a multiplier that provides the desired confidence for the coverage of the population based on the sampled size, normality, and $s_{combined}$. The quantity $s_{combined}$ includes the variability within the sample and the variability of the determination of the k_{eff} by the criticality code, i.e., the standard deviations that are provided with the individual k_{eff} values as output of the criticality calculation via the Monte Carlo code.

If normality is rejected, then the distribution free tolerance limit must be applied. The result applies to that waste type represented by that set of critical experiments.

4-15 Justify the elimination of Δk_m in the Critical Limit for Normal Distribution Tolerance Limits (NDTL).

Same argument provided for LUTB with respect to Δk_m can be applied to NDTL.

Please see the response to RAI 4-9 for DOE's position on Δk_m.

Section 4.1.3.2.3 Distribution Free Tolerance Limit

4-16 Demonstrate that the Distribution Free Tolerance Limit (DFTL) approach is at least as bounding as the lowest k_{eff} approach.

It appears that the selection of the lth k_{eff}, which is based on the l number of samples needed to provide the desired tolerance limit (e.g., 95/95) does not result in a low k_{eff} value for the critical limit. For example, based on the explanation provided by Natrella (1966, pp2-15), referenced in the topical report, using 95 critical benchmarks and 95% confidence, k_{eff} for 95% of the waste packages under a specific configuration for specific waste type will be below the third largest k_{eff} for the 95 critical benchmarks. What is needed is that with 99% confidence, 95% of the population (e.g., k_{eff}) fall below the smallest k_{eff} for the benchmark set.

The DFTL provides a Critical Limit such that for the waste form of interest, based on a sample of calculated k_{eff} values, there is a specified confidence that a stated proportion of the k_{eff} values for that population will be greater than the critical limit. The critical limit on this basis is the value that k_{eff} can not exceed without consideration of the possibility of criticality. There is high confidence that only a small portion of the population of k_{eff} values for a waste form will be below the critical limit. The critical limit is the value that divides potentially critical and non-critical classifications. In the example values of the RAI, this critical limit is the value such that there would be 99% confidence that 5% of the population of k_{eff} would be below this limit, which is an upper limit for application to assess potential criticality.

Choosing the smallest value of k_{eff} in a sample of n will provide, at a given confidence level, a proportion of the population that is above that value. If confidence and proportion of the population are chosen in advance, as in general applications, then, depending on sample size, the smallest observed k_{eff} may be more conservative than necessary.

As stated in the topical report, the sample size may restrict the confidence and/or the proportion of the population values that can be stated for the waste type when using the DFTL to determine the critical limit. If this is the case, there may be a need to obtain more k_{eff} data to provide the desired confidence and population coverage. DOE plans to reprocess any situation in which more data are obtained using the logic of Figure 4-1, for completeness.

4-17 Justify the elimination of Δk_m in the DFTL approach.

Same argument provided for LUTB with respect to Δk_m can be applied to DFTL.

Please see the response to RAI 4-9 for DOE's position on Δk_m.

4-18 Explain the use of the "3 standard deviations (3σ)" limit in a distribution-free mode.

It is not clear why the "3 standard deviations (3σ)" is used. Is 3σ enough to capture all possible scenarios?

The individual values of k_{eff} are summary values of many trials, with an associated variability that is estimated by the MCNP standard deviation. The reduction of the individual values of k_{eff} by 3σ is made to account for this variability, and these uncertainties are assumed to be normally distributed. For the normal distribution, 3σ would account for about 99.86% of the distribution, which is adequate to capture most, if not all, situations. This is the degree of conservatism offered. It cannot include all situations, and this method quantifies what is not covered.

In this situation, if the value of a specific k_{eff} is greater than 1.0, then the 3σ is deducted from 1.0, respecting the "non-positive bias" treatment. These modified values of k_{eff} are sorted to obtain the tolerance limit value that is the critical limit.

Section 4.1.3.3.1 Range of Neutronic Parameters

4-19 Provide a justification for not using a systematic approach used to identify the area and range of applicability with respect to criticality model validation for each configuration class and waste form.

The approach outlined in Section 4.1.3.3.1 is neither fully consistent with the approach in Lichtenwalter et al., nor is it comprehensive and complete with respect to identifying those parameters which may exhibit a trend in the criticality code bias.

Material concentrations, geometry, and spectrum are the areas [i.e., area of applicability [AOA]) within which the benchmarks must be evaluated for their applicability to the specific configuration class and waste form. Furthermore, there are sub-areas, if you will, within each of these AOAs which categorize the substantial variances within each of these AOAs, some of which are indicated in Page 4-18. Then, subsets of benchmarks which are based on waste package configuration class, waste form, and/or benchmark classes (e.g., Table 4.1 in Lichtenwalter et al.) need to be identified. After that, specific variables which can represent each of those categories and presence or absence of any associated statistically significant trends must be identified.

DOE intends to use a systematic, comprehensive, and complete approach to identify the area and range of applicability with respect to criticality model validation for each configuration class and waste form. DOE's approach to identifying the range of applicability for each configuration and waste form is expected to be presented in the validation reports. A planned description of the validation report is provided with this response package. In particular, Part II.A of Attachment A addresses the planned development of subsets. DOE plans to provide a basic description of the method to be used in a revision to the Topical Report.

DOE agrees that the approach in the Topical Report is not fully consistent with the approach in the Lichtenwalter et al., nor is it comprehensive or complete. It is intended that the approach used for identifying a range of applicability would be consistent with Lichtenwalter et al., although not necessarily identical. DOE plans to carefully evaluate Lichtenwalter et al. for applicability and will incorporate additional guidance from it as appropriate. Lichtenwalter et al. was intended for transportation packages, whereas the Topical Report addresses a geological repository, both of which differ in the design approach and regulations involved.

4-20 Clarify why the values for AENCF in CRWMS M&O 1998n are in mega electron volt (MEV) range as opposed to fractional or single digit electron volt.

Staff notes that AENCF inappropriately weights higher energy neutrons, resulting in thermal systems having an AENCF in the 10 keV range, whereas the predominant fission rate spectrum is actually centered in the 0.1 eV range of neutron energies. Use of Energy of Average Lethargy of neutrons causing Fission (EALF) will correct this problem. SCALE4.4 now includes the EALF parameter in its output. A corresponding Type 4 tally specification for MCNP4B can be designed by the code user.

DOE believes that discussions of specific spectrum parameters for characterizing configurations or the value ranges of the spectrum paramters are application issues and therefore not appropriate for inclusion in the Topical Report except as examples. The following discussion is provided for informational purposes only.

AENCF is an energy-times-weight tally divided by a weight tally. Therefore, the average energy calculated will be in the MeV range, even in a thermal system. Lethargy, on the other hand, is the log of the inverse of energy (Ln (10/E), where E is in MeV.). Therefore, the units of EALF will be much lower, in the fractional or single digit electron volt range and can be obtained using a type 4 tally specification using MCNP4B.

DOE agrees that the equivalent of the EALF parameter can be obtained using a type 4 tally specification using MCNP4B. As to which is more appropriate, for thermal systems the EALF seems intuitively more appropriate than the AENCF parameter. Both of these parameters will be considered. The one that is appropriate for the configurations and waste form analyzed will be chosen, assuming that either one shows the most meaningful trend when compared to other parameters investigated.

The Topical Report used AENCF as an example since a significant trend was found when using all of the laboratory critical experiments that were described in the appropriate references supplied with the Topical Report. The use of this example gives the impression that AENCF is the only spectral parameter that will be analyzed and further that it is an appropriate parameter for intact commercial fuel. Because NRC's review is not covering the additional requests made in the Topical Report, such as approval of

models, or the specific benchmark experiments, DOE will remove those parts that imply AENCF is a trending parameter or is fundamental for any waste form. This includes removal of Figure 4-3 from the Topical Report and any other information that implies a parameter has been selected for trending.

Section 4.1.3.3.3 Extension of the Range of Applicability

4-21 Provide the rationale for switching from LUTB method to NDTL method for extending the range of applicability.

The method for determining Δk_c must be based on the 99.5% of future calculations as opposed to single future calculation on which NDTL may be based. Furthermore, the margin or zone of criticality must be included in Δk_m.

This section does not change the method from the LUTB to the NDTL, if the LUTB is the basis for the critical limit. In paragraph D.2, the reference is to standard regression tolerance limits for a single value of the predictor that is beyond the ROA established. This is necessary because the LUTB is based on the range of the predictor variable, and would not be correct for the extension beyond that. The use of standard regression tolerance limits will provide a lower critical limit than would be obtained by simply extending the LUTB results. The Δk_m issue is addressed in DOE's response to RAI 4-9.

4-22 Describe the approach in establishing an additional margin when performing extrapolation beyond the range of applicability.

The report indicates that an additional margin will be added when extrapolation is extended beyond the range of applicability. However, it does not discuss the approach in establishing or quantifying this additional margin. Discussion with regard to the approach in establishing additional margin beyond the range of applicability is needed.

The approach to establishing additional margin when extrapolation is made beyond the range of applicability will depend on the nature of the bias and the applicable experiments used to establish the bias. Thus, the approach is dependent upon the waste form and its configuration, as well as various aspects of the applicable experiments. The specific approach is therefore an application issue. DOE expects to document the methodology and justification of the specific approach in the appropriate model validation report (see Attachment A). A general approach for extension of the range of applicability is described in Section 4.1.3.3.3 of the Topical Report. The following general discussion is provided for informational purposes only.

In general, there are several approaches, some or all of which may be used to perform this extrapolation. The following approach relies heavily on Appendix C of ANS-8.1-1998, and is not necessarily definitive. That is, variations of the approach given below or even other approaches not presented here may be appropriate.

> "The area (or areas) of applicability of a calculational method may be extended beyond the range of experimental conditions over which the bias is established by making use of correlated trends in the bias." (ANS-8.1-1998, Appendix C).

As stated in the Topical Report, the first step is to understand the nature of the bias. For this reason, the calculational method should be "subjected to a study of the bias and potentially compensating biases associated with individual changes in materials, geometries or neutron spectra". This will allow changes that can affect the extension to be independently validated. In practice this can be accomplished in a step-wise approach; that is, benchmarks for the validation should be chosen (where possible) such that the selected experiments differed from previous experiments by the addition of one new parameter so the effect of only the new parameter on the bias can be observed. (ANS-8.1-1998, Appendix C) In practice this may be difficult or prove to be prohibitively expensive. However, one can analyze a subset of a series of experiments and subsequently add the data from the remainder of the set of experiments to verify that indeed the bias continues to change in a systematic matter as discussed above. An understanding of the bias may also be obtained by an analysis of the physics of the series of experiments to determine if there are reasonable effects that may be causing these biases. The overall effect of these suspected parameters then may be tested by sensitivity studies that vary these parameters. If it can be shown that the sensitivity studies alter the bias, these may provide some reasonable assurance that extension of the area of applicability is reasonable and may point to a justification for the additional margin to be used.

In addition to the technique above, the calculational method should be "supplemented by alternative calculational methods to provide an independent estimate of the bias (or biases) in the extended area (or areas) of applicability." (ANS-8.1-1998, Appendix C) DOE expects that this alternative method will, to the extent feasible, be one that has been similarly benchmarked against experimental data and is established by industry practices for its intended use. To the extent feasible, it will use techniques that have been previously approved by the Nuclear Regulatory Commission for its use or that have been widely used and accepted in the industry. Further, this method is expected to be chosen to compensate for some of the known weaknesses in the primary method. The secondary method may also be more efficient, or more appropriate for sensitivity studies than a Monte Carlo method. After this analysis is performed, and with an understanding of the nature of the bias, a qualified analyst can determine a reasonable additional margin to be used for the application by a comparing the results of each method and choosing a bounding value for the additional margin.

As a supplement to, or in place of, the sensitivity studies described above, a stochastic analysis similar to the one used for the additional margin provided for the uncertainty of decay constants and branching fractions of radinuclides may be used. This latter approach is preferable if there is some doubt that an analysis is bounding over the entire range of application of the added margin.

The application of an additional margin could also consider other additional margins that are being used as a result of a particular design approach used for a specific waste form, if these margins can be defined.

An arbitrary margin could also be used. This additional margin may be arbitrarily large (e.g., 0.05) if any of the above stated approaches are unable to produce a reasonable assignment of the additional margin.

Documentation of the specific methodology and rationale for determing the additional margin is expected to be provided in the appropriate validation report.

Section 4.1.3.4 Discussion of Results

4-23 Present the results in terms of their applicability to the waste package configurations under repository conditions with respect to material, geometry, and spectrum.

Table 4-1 presents only the results of modeling and calculating k_{eff} for the Laboratory Critical Experiments (LCE) and CRC without making any connection to their applicability to the different waste package configurations in the repository with respect to specific ranges of parameters covering material, geometry, and spectrum.

DOE believes that the example results presented in Table 4-1 are part of the application information and not part of the methodology. The following discussion is preliminary and for informational purposes only.

DOE plans to provide the type of information from Table 4-1 and the presentation of the results in the manner requested in the RAI in the validation reports. Until the designs are established, the results presented in Table 4-1 cannot be presented in terms of their applicability to waste packages or repository conditions. Table 4-1 was shown to give a sense of the accuracy of the chosen method by showing average values of k_{eff} calculated and its standard deviation. It will be removed from the Topical Report, as will the requests for acceptance made in the Topical Report that are related to specific applications such as specific benchmark experiments.

Section 4.1.3.4.1 Trending Results for Commercial Spent Nuclear Fuel

4-24 Demonstrate the applicability of CRCs to the waste package configuration with the intact waste form with respect to the following areas:

(a) Material (e.g., plate boron concentration, soluble boron concentration, reflector composition, fuel material properties, etc...)

(b) Geometry (e.g., assembly separation distance, poison plate thickness, reflector wall separation distance, etc...)

(c) Spectrum (e.g., Average Energy for Neutrons Causing Fission (AENCF) compared to AENCF for intact waste form)

CRWMS M&O 1998n does not establish the applicability of CRCs to the waste package with the intact waste form as requested in this topical report. For example, Table 2.4-1 on Page 60 of CRWMS M&O 1998n shows the AENCF range for CRCs are only between 0.2475 MEV and 0.2643 MEV. However, the same table shows the AENCF range for all the configurations in the repository is between 0.0016 MEV and 0.3311 MEV. Assuming the AENCF range for the waste package configuration with intact spent fuel assemblies is somewhere between 0.0016 and 0.3311 (the report should specify the AENCF along with all the relevant benchmarking parameter ranges for the intact spent fuel assemblies), at least the CRC range with respect to AENCF spectral index must cover the waste package configuration with the intact waste form.

DOE will remove the request in the Topical Report for approval of application of the methodology to commercial fuel in an intact form and those parts that imply AENCF is the chosen trending parameter or is fundamental for any other waste form. The formal demonstration of CRC applicability will be provided in the model validation reports. (Attachment A provides the proposed planned workscope for these reports.) However, a brief response to the questions, for informational purposes only, is provided below. The source of this information is CRWMS M&O 1999.

Many parameters, in addition to those used as trending parameters, are expected to be applied to characterize both the expected repository configurations and the experiments in the waste form and configuration class specific subsets. These parameters may include, but are not limited to: cooling time, fuel temperature, fuel type, cladding type, fuel density, geometry type (e.g. square lattice, triangular lattice, homogeneous solution), absorber types, and absorber concentrations. In addition to these, DOE expects to trend against select spectral, material and geometry specific parameters. The trending parameters may include, but are not limited to: neutron spectrum ratios, enrichment, burnup, actinide ratios, plutonium concentrations, boron concentrations, fuel pellet diameter, fuel rod spacing, and fuel rod pitch to fuel pellet diameter ratios. The trending parameters may also include parameters that consider the absorption, fission and leakage spectrum. (See Attachment A, Parts II.B.2 and II.D.)

Page 179 of Lichtenwalter et al. (NUREG/CR-6361) states that three fundamental parameters should be considered in the selection of suitable experiments for use in the evaluation of transportation and storage package designs. They are as follows: (1) geometry of construction, (2) materials of construction (including fissionable material), and (3) the inherent neutron energy spectrum affecting the fissionable material. The following discussion provides a general qualitative description of the characteristic ranges that influence the neutronic behavior in the CRCs and waste package, and the applicability of the CRCs to a waste package. It should be noted that the CRCs are included as part of the establishment of bias and uncertainty for a critical limit, which is used in the generation of a loading curve. As stated in the Topical Report, laboratory

critical experiments and radiochemical assays will also be used. The CRCs are not assumed to fully correspond with the waste package configuration, which is why the LCEs are analyzed (i.e., to consider absorber plate effects, reflector materials). Figure 4.24-1 is a representative comparison of the relative neutron spectrum in the waste package and in a CRC. This waste package configuration was composed of burned fuel ranging in assembly average burnup from 16.358 GWd/mtU through 34.416 GWd/mtU, with an essentially infinite water reflector.

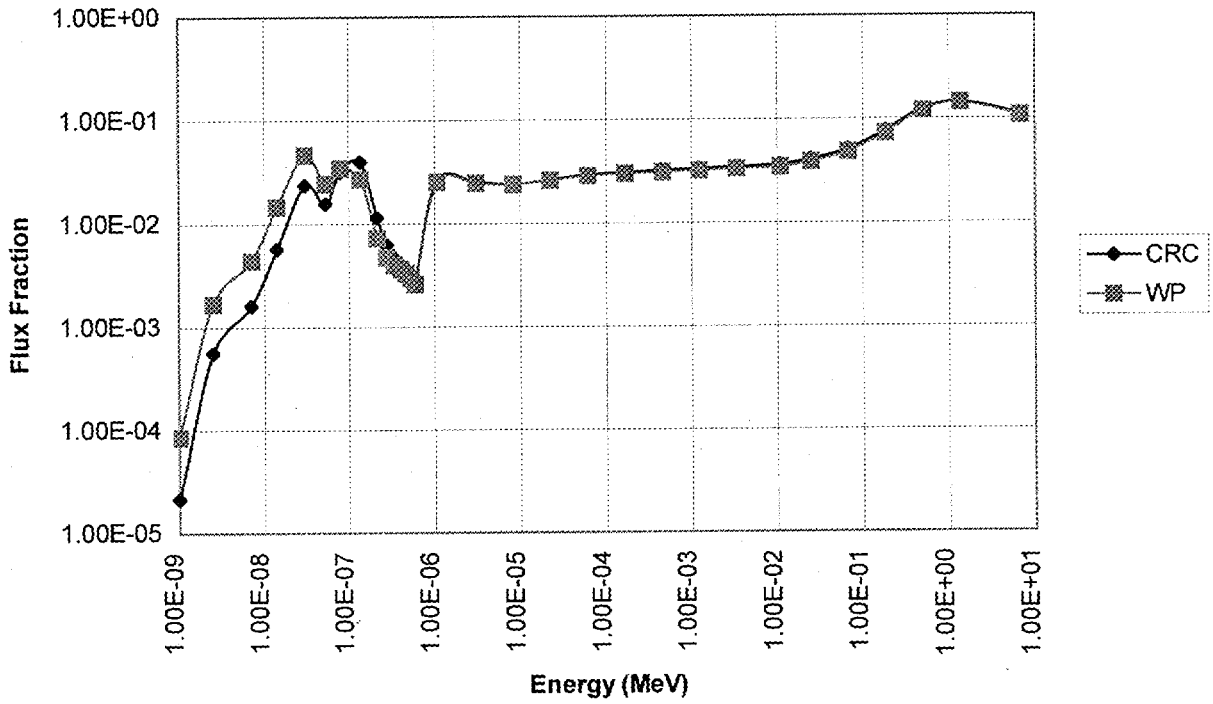

Figure 4.24-1. Neutron Spectra for CRC Core and Inner Cavity of waste package

As can be seen in Figure 4.24-1 the relative flux spectra are essentially identical in the CRCs and waste packages above 1.0 Mev and between 4 eV and 1.0 MeV. Figure 4.24-1 shows that the CRC relative flux spectrum is lower at energies below 0.1 eV than the comparable waste package. This behavior is expected from the differences in temperature and fuel-to-moderator ratios. This slight shift in spectrum has a small effect on the fission reaction rate. The effect of this shift in neutron spectrum is expected to be accounted for by the waste package design (e.g., use of neutron absorbers). Thus, the neutron spectra in the waste package and in the CRC are very similar.

CRWMS M&O 1999. *Waste Package, LCE, CRC, and Radiochemical Assay Comparison Evaluation.* B00000000-01717-0210-00107 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990812.0351.

Section 4.2.1 Corrosion Model

4-25 Justify the extensive reliance on the wide range of corrosion rates utilized to determine the probability and location of a WP breach.

Given sufficient criticality control in the as-fabricated WP, a breach in the WP is necessary for a criticality event to occur. The model used to determine the probability of a WP breach and its location was the WAPDEG code using the Total System Performance Assessment (TSPA)-Viability Assessment (VA) base case. The primary limitation of this case is that the input parameters for corrosion rate rely extensively on expert elicitation, with nearly five orders of magnitude variance in the corrosion rate utilized. Thus, the possibility exists for a wide range of WP failure times and a commensurately wide range of times in which criticality control becomes important. The wide range of corrosion rates resulting from the heavy reliance on expert elicitation is considered a limitation to the utility and validity of the subsequent criticality analysis, since it leads to dilution of the probability of occurrence and resulting risks.

The expert elicitation process for estimating corrosion rates for the more corrosion-resistant steels was only a temporary expedient, caused by the lack of corrosion test data, for preliminary analyses. The range of possible corrosion rates is expected to be considerably narrowed by the multi-million dollar corrosion testing program that is underway at Lawrence Livermore National Laboratory and the University of Virginia. The M&O quality assurance program will ensure that the data from these programs will be properly translated into corrosion rates that will strengthen the criticality probability estimates. The Topical Report will be revised to state that corrosion rates are expected to be determined from an extensive corrosion testing program.

4-26 Provide justification for the long-term credit being taken for the presence of fuel cladding in the degradation analysis.

It appears that credit is being taken for the presence of Zircaloy-4 cladding in terms of its corrosion resistance. Although Zircaloy-4 does have good corrosion resistance, it is known to suffer from localized corrosion under reasonably attainable conditions inside breached WPs. Additionally, the cladding can be degraded prior to disposal due to the effects of irradiation, reactor water chemistry, and predisposal storage conditions. Commercial SNF exhibits a wide range of Zircalloy material characteristics, including large variations in the degree of hydriding, oxidation, erosion thinning, embrittlement, crack formation, pellet-cladding interactions, crud depositions. Further information is requested on the technical basis that this degree of credit can be claimed for the Zircaloy-4 cladding and the effect on criticality control if no credit is taken.

DOE believes that the inputs used in the degradation models are an application issue and therefore not appropriate for inclusion in the Topical Report. The following dicussion is provided for informational purposes.

Future TSPAs and criticality evaluations used to support the License Application are expected to have a common approach to Zircaloy cladding credit for commercial SNF that will be based on the extensive testing, carefully reviewed modeling, and appropriate conservative assumptions.

Section 4.2.2 Internal Geochemistry Models

4-27 Clarify the internal geochemistry model treatment of Uranium (U) produced from WF degradation.

WF U is input into solution (along with other degradation products) according to fixed degradation rates and solution evolution modeled in EQ6. What is not clear is if U secondary phases are allowed to precipitate, in effect lowering the U release rate and perhaps lowering the probability for potentially critical external accumulations. References cited in the topical report suggest that secondary U phases will be included in internal degradation models. These references include CRWMS M&O (1998e)[7] and CRWMS M&O (1998q)[8], the latter being cited in the former. For example, in section 6.3.2 of CRWMS M&O (1998e), EQ6 models of SNF degradation are said to lead to precipitation of the hydrated uranyl silicate soddyite. In contrast, retention by secondary U phases are not modeled in the TSPA-VA. Clarify whether these differing approaches will be reconciled in future work.

The observations of the staff are correct. The references cited in the Topical Report consider the precipitation of secondary phases, while TSPA-VA does not. There are two reasons for the difference in approach. First, there is a difference between the conservative objectives. For criticality internal to the waste package, the conservative approach allows the most neutron absorber to be removed from the waste package, while keeping the most fissile material in the waste package, hence the precipitation of uranium minerals. In contrast, TSPA is concerned with the release of radionuclides such as actinides, which include uranium. Of course, most of the uranium has very low radioactivity, so its precipitation in the waste package would have little effect on dose at the accessible environment.

This leads to the second reason for the difference in approach. Because of the strong concern with the fissile material remaining in the waste package after waste form dissolution, a steady-state flow-through capability (described in Section 3.1.1 of the Topical Report) has been added to EQ3/6 for use in the studies cited in the Topical Report. This capability was not available in 1997, when all the inputs to TSPA-VA were finalized. The geochemistry treatment of waste form uranium and other materials will be

similar in the future for both types of evaluations. DOE plans to explain any differences in the appropriate License Application document.

4-28 Compare and contrast the approach to modeling to be used for release and internal geochemistry in the criticality analysis with that employed in present and future TSPA models (which may or may not use EQ3/6).

It was stated in topical report section 4.2.1 that WF degradation modeling in the criticality analysis will employ TSPA model approaches, but it is not clear if this extends to modeling release from WPs. TSPA-VA did not explicitly employ EQ3/6 geochemical modeling of WF alteration, RN release, and secondary solid phase formation. Any deviations from the TSPA-VA approach in the criticality analysis should be demonstrably more conservative or supportable.

As stated in the response to the previous RAI, all future geochemistry calculations for performance assessment and criticality evaluations will be consistent as appropriate. They are expected to both use the most up-to-date computer codes available. The current up-to-date geochemistry code is EQ3/6, which is expected to soon be supplemented by PHREEQC. Any revision to the Topical Report will include information about PHREEQC.

4-29 Specify what kinetic models will be used in the internal geochemistry models and clarify whether default EQ6 values will be used.

Selection of kinetic models profoundly influences model results regarding degradation products and water chemistry. Cited documents discussing EQ6 degradation models (CRWMS M&O, 1998e, 1998q, and Appendix C reference 1998e) do not address kinetic models affecting the rates at which WP and WF degradation products precipitate. It appears that either default EQ6 kinetic parameters are utilized or kinetics are not included. Because degradation products are integral to criticality models, calculations predicting their formation should rely on supportable or conservative kinetic data. This comment applies also to the external geochemistry models discussed in section 4.2.3 of the topical report.

EQ6 calculations are run in kinetic mode (i.e., not in the reaction progress mode). Most calculations use constant rates (EQ6 input parameter nrk=3) for the degradation of initial solid components, with the rate specified in the input file parameter rk1, in moles/(cm^2·sec). The constant rate is used because (1) limited rate data are available for corrosion of most package materials, and (2) most waste package reactants are grossly out of equilibrium with the aqueous component of the system, and there is no benefit to tracking the affinity of the waste package "reactants". In calculations currently underway, true transition state theory (TST) rates, taken from the literature, are used for modeling silicate reactions with package effluents.

To ensure conservatism, many rate combinations are used for models of waste package degradation. In current calculations, the rk1 of each package material may take two to three values, covering a range of 1 to 3 orders of magnitude. Each EQ6 run combines average-high or low-high-average rates for glass, fissile material and steels, with drip rates (water influx rates) ranging from 0.0015 to 0.5 m^3/year. Combinations of low rates from one material, and high rates from another, may produce the most "conservative" results; e.g. a low glass rate may allow acidity to build from rapid steel degradation, increasing loss of gadolinium. While this "rate matrix" approach requires large numbers of runs, it is expected to uncover unusual conditions that may enhance solubility or precipitation of fissile materials.

For most of the degradation products the kinetic rule is instant equilibrium. However, for those phases for which a kinetic model is controlling (particularly silica bearing minerals), the transition state theory kinetic model is used.

4-30 Justify the use of J-13 well water as representative of the solution that would be present within the WP.

Once the WP is breached, corrosion and degradation of WP internals play important roles in criticality control. Although some testing data has been obtained, the results are based on experiments conducted in variants of simulated J-13 well water. It seems unlikely that the chemistry of the solution inside a WP would be a J-13. Furthermore, it is unclear if the testing program and the subsequent analysis has considered the possibility of chemistry changes resulting from evaporative processes and dissolution products from WP components (e.g., acidification due to metal cation hydrolysis, alkalinization from dissolution of HLW glass, etc.). Because of possible chemistry change, the corrosion mode and corrosion rates could be altered from the general corrosion case considered. For example, alkalinizaion could lead to the formation of a passive film on carbon steel components that could then experience localized corrosion in the form of pitting or crevice corrosion in the presence of chloride. Similarly, the corrosion mode of stainless steel components could change from relatively slow passive dissolution to more rapid localized corrosion, which could lead to unanticipated, catastrophic failure of WP internal components. These accelerated corrosion modes could make conditions for criticality more favorable by allowing fuel materials to coalesce. Further information justifying the environments chosen and any further work examining likely alternate chemistries and their effects on material degradation is requested.

J-13 water has been used for the composition of water entering the waste package, but never for the composition of the water *in* the waste package. A principal objective of the geochemistry code calculations is to determine what aqueous composition is expected to result from the various degradation processes. The composition of the solution in the waste package is a function of the volume of standing water in the waste package and the rate of flow into (and out of) the waste package. The geochemistry calculations cover a range of values for these parameters (CRWMS M&O 1998a).

DOE believes that the inputs used in the geochemistry and degradation models are application issues and therefore are not appropriate for inclusion in the Topical Report. The following information is preliminary and is provided for informational purposes.

The sensitivity of geochemistry results to the composition of the *incoming* J-13 water is being tested by varying the principal parameters over a set of likely ranges. These parameters are determined by DOE's geochemistry experts. In general, the waste package degradation calculations (CRWMS M&O 1998a) predicted that the greatest release of uranium, plutonium, and gadolinium (which is very important because it is an added neutron absorber for disposal of several of the more highly reactive waste forms) occurs when there is rapid degradation of one or more waste package components. Rapid degradation of waste package components has a larger effect on the waste package aqueous chemistry than does any possible variations from the incoming J-13 water composition (CRWMS M&O 1998b). For example, for those waste forms involving codisposal with HLW glass, the rapid glass degradation will often drive the ionic strength to ~1 molal. Steel degradation somewhat faster than normal can drive pH below 5. Under these conditions, the aqueous phase deviates from the nominal J-13 composition by much more than the variety of suggested modifications for the composition of the indripping water.

Any licensing applications of geochemistry codes will be supported by a thorough sensitivity analysis with respect to composition of the indripping water. The following is typical of the sensitivity analyses performed thus far. DOE varied the equilibrated $\log_{10}(fCO_2)$ values from –2.5 to –3.5, and the Ca^{++} concentration constraints were also varied. The calculated uranium loss from the package varied by a factor of <3 (principally due to fCO_2 variations), but the calculated gadolinium loss varied by less than 20%.

With respect to the last two sentences (accelerated corrosion rates) of this RAI, the Topical Report is expected to be modified to indicate that if a geochemistry evaluation shows very low pH, or other corrosion-enhancing condition, that geochemistry is expected be re-evaluated with appropriately enhanced corrosion rates reflecting for the affected waste package components.

CRWMS M&O 1998a. *EQ6 Calculations for Chemical Degradation of PWR and MOX Spent Fuel Waste Packages*. BBA000000-01717-0210-00009 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19980701.0483.

CRWMS M&O 1998b. *EQ6 Calculations for Chemical Degradation of Fast Flux Test Facility (FFTF) Waste Packages*. BBA000000-01717-0210-00028 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19981229.0081.

Section 4.2.3 External Geochemistry Models

4-31 Describe how colloidal deposition will be incorporated into modeled chemical deposition.

Indicate whether the approach will be the same as those adopted under TSPA. Evaluation of models of fissile material accumulation requires full understanding of colloid modeling. A previous analysis of external criticality (CRWMS M&O, 1998p) concluded that colloidal transport and accumulation of fissile materials would be insignificant. It should be clear how new analyses will differ and to what extent they are supportable and conservative.

Criticality evaluations are expected to use the same model for colloids (formation, transport, and adsorption) as does TSPA. The Topical Report will be revised to state this. This model is currently under development and will be used as soon as it is available.

Section 4.2.4.1 Validation of Degraded Methodology

4-32 Provide more information on validation methods for the "pseudo flow-through" internal and "open system" external EQ6 models.

With regard to the "pseudo flow through" model, topical report section 4.2.4.1 refers only to hand calculations supporting the solute concentration adjustments (CRWMS M&O, 1998q). This exercise only partially addresses the question of the validity of the model results.

In discussing the "open system model," it is stated that the results are conservative, but the pertinent reference (CRWMS M&O, 1997f) is missing from the chapter 6 reference list. The report acknowledges that validation has not yet been done, but does not describe how it will be done. This information is vital to assessing the methodology (see also discussion of topical report section 4.2.4.2 below). Validation approaches should provide confidence that models will not underestimate the effects of processes that could lead to criticality.

The pseudo flow-through mode of EQ3/6 mentioned in the Topical Report has been replaced by a modification to the EQ3/6 code that enables the modeling of water inflow and outflow to track the timestep adjustment process exactly, thereby ensuring not only that the chemical changes are accurately resolved in time, but that they also accurately reflect the volume of water in the waste package at any given time. This new version of EQ6 had been qualified (CRWMS M&O 1998, CRWMS M&O 1999a , CRWMS M&O 1999b). The Software Qualification Report for this new version of EQ6 (CRWMS M&O 1998) includes tests of the solid-centered flow-through method. The tests include comparisons against analytical solutions, and also comparisons against results obtained by chaining several thousand individual EQ6 runs (with adjustment of the water mass between each run). The agreement among the different methods was quite good.

With respect to the open system model discussed in the Topical Report, it should be mentioned that DOE has added the geochemical transport code, PHREEQC, to our tools library. The open system external model discussed in the Topical Report was a temporary expedient, and has been replaced by the industry standard geochemical transport code, PHREEQC, which is expected to be supplemented with a new version of EQ3/6 incorporating a Lagrangian transport model. The validation of these codes is discussed in the response to RAI 4-33, below.

The Topical Report will be revised to reflect the additional information presented in this item, including a discussion of the additional geochemical code.

CRWMS M&O 1998. *Software Qualification Report (SQR) for Addendum to Existing LLLNL Document UCRL-MA-110662 PT IV: Implementation of a Solid-Centered Flow-Through Mode for EQ6 Version 7.2b.* CSCI:UCRL-MA-110662 V 7.2b, SCR: LSCR198, MI: 30084-M04-001. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990312.0336

CRWMS M&O 1999a. *Software Qualification Report (SQR) for Addendum to Existing LLNL Document UCRL-MA-110662 PT IV: Implementation of a Solid-Centered Flow-Through Mode for EQ6 Version 7.2b.* CSCI:UCRL-MA-110662 V 7.2b, SCR: LSCR198, MI: 30084-M04-001. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990920.0169

CRWMS M&O 1999b. *Software Change Request (SCR) LSCR198; Addendum To EQ6 Computer Program for Theoretical Manual, Users Guide, and Related Documentation* UCRL-MA-110662 PT IV (C). Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990305.0112.

Section 4.2.4.2 Validation of the EQ3/6 Geochemistry Code

4-33 Provide additional information on the validation of EQ3/6 for the specific applications. The validation examples provided in topical report section 4.2.4.2 (Bourcier 1994[9], Bruton and Shaw 1988[10], Bruton, 1996[11], Wolery and Daveler, 1992[12]) do not adequately cover the conditions and processes to be included in the models. For example, the validated spent fuel and HLW models (Table 4-3) did not include the other waste package components (e.g., metal plates) to be included in the internal models. In addition, no examples are given that are comparable to the external models of low-temperature interaction between drift effluent waters and fracture walls. The DOE should state whether or not any new analyses will be performed that would support validation under the conditions to be modeled and, if not, how model confidence will be improved.

As the RAI notes, the validation discussions in the Topical Report were examples and not intended to fully validate EQ3/6. DOE believes that full validation is an application issue and therefore not appropriate for inclusion in the Topical Report. Additional validation

of the EQ3/6 code will include two activities not used previously: (1) comparison with alternative analyses that implement conservative approximations for processes lacking experimental verification. (2) comparison with the geochemistry/transport code PHREEQC. This latter code is particularly important because it brings some much needed modes for transport and adsorption in the invert and rock. In addition, new versions of EQ6, under validation, include radioactive decay (particularly conversion of Pu-239 to U-235) and a legitimated Lagrangian transport capability. Full validation of EQ3/6 is expected to be provided in the validation reports and referenced in the supporting documents for the License Application.

Transport calculations, similar to those in the Topical Report, may be repeated by two codes (e.g., PHREEQC and linked EQ6 runs, or PHREEQC and the Lagrangian version of EQ6), if there is reason to believe that it will enhance the credibility of the results. The qualification of these codes involves comparison against each other, against more complex analytical solutions for reactive transport, and against experimental data (both laboratory and natural analogs). With regard to the latter, the following are typical examples.

Soler, J.M. and Lasaga, A.C. 1998. An Advection-Dispersion-Reaction Model of Bauxite Formation. Jour. Hydrology, 209, 311-330.

Steefel, C.I. and Lichner, P.C. 1998. Multicomponent Reactive Transport in Discrete Fractures: I. Controls on Reaction Front Geometry. Jour. Hydrology, 209, 186-199.

Steefel, C.I. and Lichner, P.C. 1998. Multicomponent Reactive Transport in Discrete Fractures: II. Infiltration of Hyperalkaline Groundwater at Maqarin, Jordan, a Natural Analogue Site. Jour. Hydrology, 209, 200-224.

The three examples above are relevant for several reasons. First, the time scales of reaction approach those expected for degradation of the waste packages (10^2 to 10^6 years). Second, the examples involve slow flow rates and Peclet (Pe) and Damköhler (Da) numbers similar to those expected for transport through the fractured tuff. Third, many of the waste package scenarios that yield highest potential for actinide deposition involve penetration of fractures with hyperalkaline fluids (the subject of one of the Steefel and Lichner test cases). Fourth, the solution method developed by Steefel and Lichner is closely tied to the LaGrangian method currently under development, so it should be straightforward to translate the Maqarin example into a validation case.

It should be noted that natural analogues are not necessarily the strongest tests, because the timing and rates of fluid infiltration are poorly constrained. In fact, for natural analogues, the hydrological parameters are often treated as free parameters. As validation examples, they are principally useful in showing that reasonable mineral assemblages can be achieved, and for benchmarking codes based on different algorithms.

Despite the difference in time scales, laboratory tests involving reaction and deposition in flow columns often provide more conclusive tests for benchmark purposes, because flow

rates and surface areas are well-known. As long as the Damkohler number (Da) and Peclet number (Pe) are reasonably close to those projected for near-field reactive transport, and the tests involve similar reactive mechanisms (e.g., pH and CO_2 controls by reaction with tuff silicates), short-term experiments can serve as adequate tests. Several experiments underway at Lawerence Livermore National Laboratory show promise as code benchmarks. These experiments involve reaction between J-13-like fluids, crushed geomaterials, and actinides or radionuclide simulants. The progress of these experiments will be tracked, and the results will be considered for our benchmark suite in the validation reports.

Discussion of the additional geochemical codes will be added to the revision of the Topical Report.

4-34 Provide additional information on the validity or conservatism of geochemical parameters to be used in EQ3/6 models.

As acknowledged in this report, there are large uncertainties in thermodynamic and kinetic data used by EQ3/6. The report states that a range of reaction rate values will be used so that conservative cases may be identified. This analysis should take account of any synergistic effects of varying rates for the numerous solid phases involved in this complex system. Such analysis should also be applied to thermodynamic data, particularly with regard to actinide phases. (Note, for example, that much uncertainty exists regarding appropriate thermodynamic data for U and Pu phases.) Only in this way can the model results be interpreted with confidence.

The risk-informed approach to uncertainties in parameters influencing the occurrence of a criticality is to perform sensitivity analyses covering the range of possibilities for these parameters and map the results into probability distributions. The resulting probability distributions can then be used, with *conservative* estimates of consequences, to develop a conservative estimate of risk. Sensitivity of EQ3/6 calculations to uncertainties in kinetic models, and to uncertainties in the composition of incoming water, has been discussed in the responses to RAIs 4-29 and 4-30, respectively. The response to this RAI discusses recent studies of the sensitivity to variations in thermodynamic data to be used in EQ6. The sensitivity studies have all been with respect to one parameter, or one paremeter set, at a time. For those variations that have the possibility of synergistic effects, DOE plans additional sensitivity comparisons with multiple parameter variations.

When there is uncertainty in the solubility of a fundamental phase, such as $PuO_2 \cdot 2H_2O$ ($\equiv Pu(OH)_4$), the $\log_{10}K$ of the phase is varied in the EQ6 (or other geochemistry code) reference files, to determine the total system sensitivity. Figure 4.34-1 shows a comparison of gadolinium (Gd) solubility for two alternative thermodynamic datasets, SKB and Weger (Spahiu and Bruno 1995, Weger et al. 1998). Also shown in this figure is a comparison of the concentration of the controlling Gd mineral phase, gadolinium carbonate, for the two alternative datasets. There is a significant difference in concentration (between the two alternative datasets) over most of the time. However,

during time of peak Gd concentration, and minimum gadolinium carbonate concentration, the two datasets give identical results.

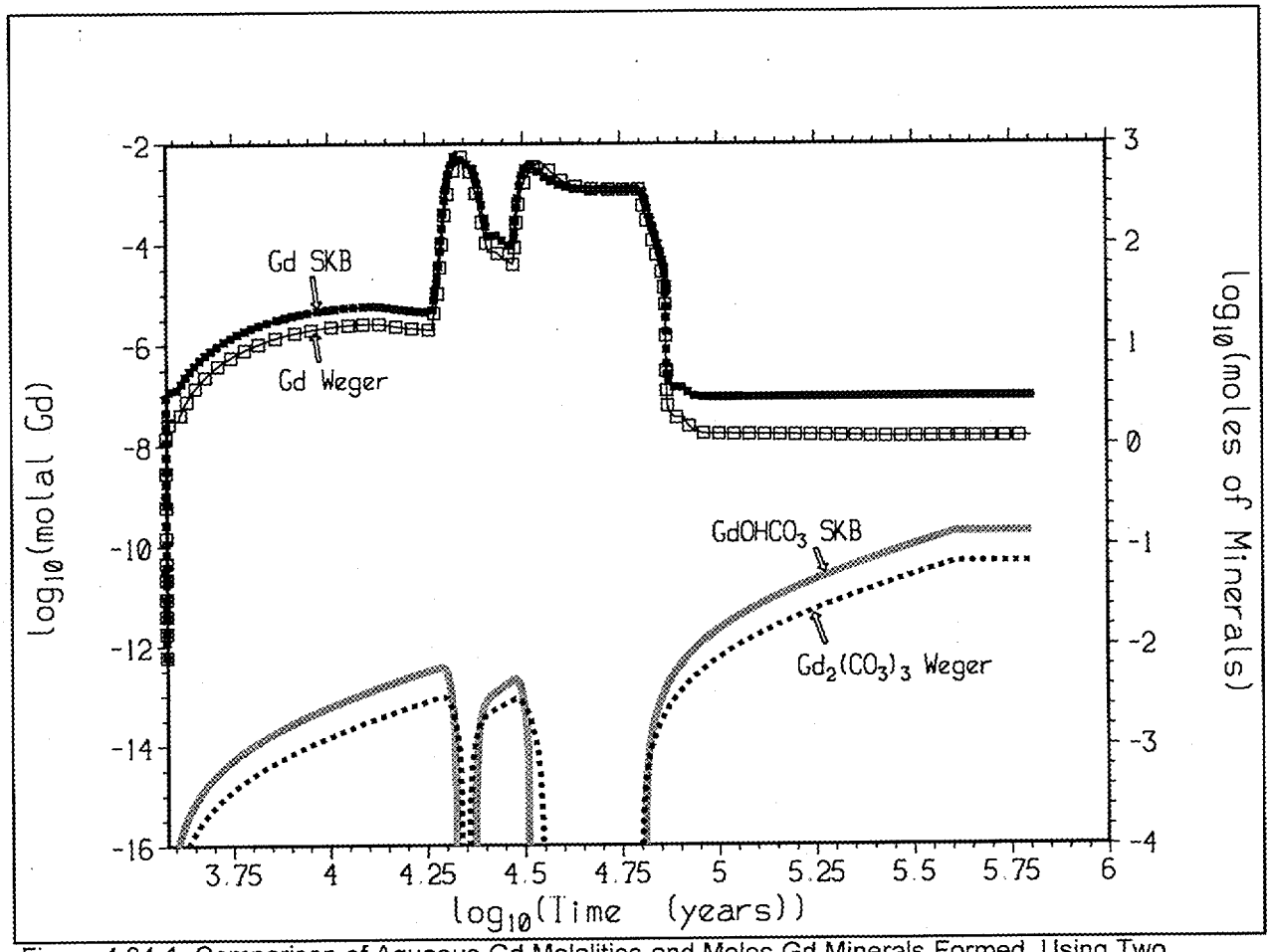

Figure 4.34-1. Comparison of Aqueous Gd Molalities and Moles Gd Minerals Formed, Using Two Different Sets of Thermodynamic Constants. (CRWMS M&O 1999, Figure 5-5)

DOE believes that the parameter of greatest interest to criticality is the loss of Gd from the waste package. This value is equal to the integral over time of the product of the Gd concentration in solution multiplied by the volumetric flow rate out of the waste package. In all the cases analyzed that show significant Gd loss, most of the loss is found to occur during the peak aqueous Gd concentration (CRWMS M&O 1999, *Waste Package Related Impacts of Plutonium Disposition Waste Forms in a Geologic Repository,* Section 6.2.3.1). Therefore, since the SKB and Weger thermodynamic datasets give nearly identical peak aqueous Gd concentrations, the difference in Gd lost from the waste package is expected to be small. This comparison is given in Table 4.34-1. In this table a comparison between the two sets of thermodynamic constants is presented for two examples, which are labeled Run 4 and Run 6, and which have representative, but different, conditions. It

can be seen from the table that use of the SKB database was slightly conservative, giving a slightly greater loss of Gd from the waste package for both runs.

Table 4.34-1. Percent Loss of Gadolinium, For Entire waste package, Thermodynamic Data Sensitivity Study (CRWMS M&O 1999, Table 5-6)

	Run 4	**Run 6**
SKB database	14.8625	12.9599
Weger database	14.4983	12.9537
Fractional Difference	0.024	0.00048

CRWMS M&O 1998. *Two (2) Data Cartridges for Electronic Media for EQ6 Calculations for Chemical Degradation of Fast Flux Test Facility (FFTF) Waste Packages, with Supporting Documentation.* BBA000000-01717-0210-00028, REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19981229.0082.

CRWMS M&O 1999. *EQ6 Calculation for Chemical Degradation of Pu-Ceramic Waste Packages: Effects of Updated Materials, Composition and Rates.* CAL-EDC-MD-000003 REV 00B. Las Vegas, Nevada: CRWMS M&O.

Spahiu, K. and Bruno, J. 1995. *A Selected Thermodynamic Database for REE to be Used in HLNW Performance Assessment Exercises.* SKB Technical Report 95-35. Stockholm, Sweden: Swedish Nuclear Fuel and Waste Management Co. TIC: 225493

Weger, H.T.; Rai, D.; Hess, N.J. and McGrail, B.P. 1998. *Solubility and Aqueous-Phase Reactions of Gadolinium in the K^+–Na^+–CO_3^{2-}–OH^-–H_2O System.* PNNL-11864. Richland, Washington: Pacific Northwest National Laboratory. TIC: 242377.

Section 4.3.1 Probability Concepts

4-35 Indicate how correlations between sampled parameters will be identified,

quantified, and accounted for in the criticality configuration generation codes.

Use of the Monte Carlo method requires that correlations between sampled parameters are taken into consideration if they are not truly independent variables. For example, the drip rate onto the package may affect the WP lifetime. Failure to account for these correlations could result in erroneous results.

The Monte Carlo process will always represent correlations between parameters by using appropriate conditional probability distributions for parameter sampling. For the Monte Carlo example presented in Appendix C of the Topical Report, the waste package lifetime was sampled from a probability distribution abstracted from another Monte Carlo analysis, which used the waste package degradation code, WAPDEG. The WAPDEG code used a waste package corrosion model that captured the strong negative correlation between drip rate and waste package lifetime (specifically time to first penetration). This dependence was explained in the Topical Report, Section 3.5, item #1. Although the exposition of the configuration generator in Section 4.3 of the Topical Report indicates that the drip rate is sampled (Section 4.3.2, subheading Internal Criticality, item #1), it is only used for determining the rate of removal of dissolved species (item #5 of the same subheading). Because the drip rate is not used as a determinant of the penetration time, its sampling is not in direct conflict with the completely independent sampling of the waste package barrier penetration time. There may, however, be some indirect conflict because the consequences of fissionable species removal may depend on the time since emplacement (e.g., how much Pu has decayed to U). Such correlations are expected to be better understood and captured in the Monte Carlo configuration generator for License Application.

Section 4.3.2 Monte Carlo Technique

4-36 Justify the assumption that it is acceptable to only consider the potential for one external criticality for a given realization.

DOE argued that the small probability of a realization yielding a critical configuration obviates the need to analyze the realization for multiple criticalities. This argument is acceptable only if each criticality is an independent event. Since having a single criticality in a realization requires that several sampled parameters are favorable to produce a criticality, additional criticalities are not independent events and the probability of having multiple criticalities for a single realization may not be small enough to be ignored. Failure to consider the potential for multiple criticalities in a realization may lead to an underestimation of the probability of a criticality event occurring.

The example applications of the methodology in Appendices C and D of the Topical Report considered multiple criticalities as independent events. This is not conservative if the occurrence of one criticality increases the probability of additional criticalities. The

strongest example of such positive correlation is the common mode failure. The DOE plans to evaluate all such possibilities are noted in the response to RAI 1-2.

4-37 Justify the exclusion of water chemical parameters other than pH in regards to item A of the External Criticality list on page 4-39.

Water chemistry will be greatly altered during Waste package and WF interaction, and concentrations of other components such as carbonate influence the geochemical behavior of U and Pu.

The list of sampled parameters in the first sentence of Item A on page 4-39 was only meant to be illustrative, not exhaustive. The next to last sentence of that paragraph alludes to solution characteristics in general, which would include the concentrations of all chemically significant elements (not just the fissile ones listed in the first sentence) and solution parameters (e.g., ionic strength, eH). This will be clarified in the revision to the Topical Report.

4-38 Clarify how the path selection process does not constitute an additional, non-conservative reduction in probability for a given configuration.

In item B of the External Criticality list, it is stated that random selection of external pathway is weighted according to probability. Subsequent transport modeling utilizes probability sampling of parameters. It should be made clear that this approach does not constitute redundant application of probability screening of external pathway.

The essence of the Monte Carlo process is the probability sampling of parameters that are used in the calculation of the result parameter (typically k_{eff} for our cases). The statistical summary of many Monte Carlo repetitions is used to generate the probability distribution for these result parameters. Any allusion to a probability of a pathway, scenario, or configuration is really to a probability distribution calculated in this manner, so there is generally no redundant or external source of probability calculation. These points will be clarified in the Topical Report. The principal exceptions to this rule are for the waste package time to breach and bathtub duration. The probability distribution for these parameters is abstracted from the official series of WAPDEG runs performed for the performance assessment evaluations. The example criticality probability calculation in Appendix C of the Topical Report showed that this can be accomplished, while preserving independence of the two domains.

Section 4.3.3 Configuration Generation Code

4-39 Regarding item II.C of Section 4.3.3 on the invert configuration generation code (CGC) geochemistry modeling, justify the exclusion of water chemical parameters other than pH in computing solubility dependence.

In the WP CGC, solubility dependence on other species such as carbonate is included (item I.D.). Such dependence - which, for example, is strong for carbonate content in computing U solubility-should also be included in external cases.

As explained in the response to item 4-37, there was no intention to exclude any items not listed in the first sentence of Item A on pg 4-39. The text will be clarified in the revised Topical Report.

4-40 Clarify how matrix-fracture distribution of water below the WP is calculated (item III.B and III.C). Reconcile the distinction between fracture and matrix travel times discussed in section 4.3.3 with the attribution of all flow to the fractures apparent from the discussion in section 4.2.3.

The distinction between matrix and fracture flow has profound implications for modeled travel times and water-rock interaction. For example, it is typically assumed that solutes are not sorbed during fracture flow. The distribution of groundwater flow between the fracture and the matrix will strongly affect U and Pu transport because of contrasting sorption and groundwater travel times. U and Pu transport rates and concentrations are central to models of external criticality.

The allocation of transport to the fractures gets the fissionable material to the potential accumulation zone sooner. The earlier potential criticality is conservative generally, and specifically in minimizing the decay of plutonium. The methodology actually implemented is expected to include the capability to calculate flow through the matrix and diffusion from fractures into the matrix. DOE expects to use the geochemical transport code, PHREEQC, for this purpose. A clarification will be made to the text in the revised Topical Report.

Section 4.4.1.1 Steady-State Criticality

4-41 Justify the methodology used for analyzing the steady state criticality condition with no iron oxide.

One of the reasons provided in the "Second Waste Package Probabilistic Criticality Analysis: Generation and Evaluation of Internal Criticality Configurations" report for not including no-iron oxide or no-B-10 configurations in the analysis was that the corresponding k_{eff} values are "below any possible range of linearity." This is a questionable basis for excluding these types of realistic configurations. Knowing that taking credit for boron retention in the iron oxide has been dismissed in a later report, it is very possible that the combination of a high acidic environment would cause most of the iron oxide to be dissolved and flushed out of the waste package.

The steady-state criticality with *no* iron oxide was not evaluated because it is an extremely unlikely configuration. DOE plans to evaluate the effect of reduced iron oxide concentrations and expects to provide a basis for any decision to use a minimum iron oxide concentration for evaluation. DOE does not expect that reduced iron oxide concentrations will have any effect on the consequences (increased radionuclide inventory) for the steady-state criticality, but it could effect potential insertion rate mechanisms for the transient criticality.

Section 4.4.1.2 Transient Criticality

4-42 Provide an analysis for the seismic event using the time scales such as 0.3 seconds for reactivity insertion as part of the transient criticality analysis.

The cited reference (CRWMS M&O 1997e) does not provide the transient criticality analysis with a duration of 0.3 second as implied in the topical report. The report uses 30 seconds, which is based on the terminal velocity of iron oxide particles, for the duration of reactivity insertion. However, reshuffling of spent fuel in a time duration of one second or less as the result of the seismic event with no iron oxide must be considered.

Although the reference cited described a concept that could support a 0.3 second insertion rate, subsequent analysis for intact commercial SNF assemblies showed that event to be incapable of producing a significant increase in k_{eff}, as was explained in the response to RAI 3-19. As was also explained in the response to RAI 3-19, DOE plans to evaluate the possibility of such a transient criticality for a waste package with a limited number of collapsed criticalities. This evaluation is expected to include the probability of the specific partially collapsed configurations required to support such a transient criticality.

As an extra measure of conservatism, DOE plans to re-evaluate this question with a reduced amount of iron oxide present (but not zero, since that would not be physically possible). DOE expects to provide a basis for any decision to use a minimum iron oxide concentration for evaluation.

4-43 Justify the transient criticality analysis using a computer code which does not have the restrictions that are associated with RELAP5/MOD3.

The one-dimensional RELAP computer code has been developed for reactor cores which have flows parallel to the fuel bundles. The code is not intended to be used for systems with cross flows of more than 10%. First, the validity of using a one dimensional code for two dimensional analysis is not demonstrated. Secondly, the flow in both dimensions in the waste package model are perpendicular to the fuel assemblies in which the RELAP has not been designed. Thirdly, no benchmarks which would demonstrate the degree of applicability and accuracy of RELAP5/MOD3 for the waste package transient criticality conditions, are offered. Other codes which have the capability to perform three dimensional

thermal hydraulics analysis might be the more appropriate computer code to use for analyzing the waste package transient conditions.

It is recognized that the primary hydrodynamic flow in RELAP5 is along the fuel axis. DOE has overcome this limitation by developing RELAP models for components that represent the cross-flow, as explained in Section 4.4.3.2 of the Topical Report. Any licensing presentations of this sort of analysis are expected to demonstrate accuracy by comparison of the behavior of the component models with actual test data.

4-44 Justify the use if computer codes that do not have temperature feedback capability to determine the reactivity of these waste package systems.

The approach proposed in CRWMS M&O 1997e with regard to compensating of the lack of the temperature feedback in MCNP4A, due to unavailability of "an associated cross section library with sufficient temperature data to calculate reactivity changes," does not appear to be very sound. The use of SAS2H, modified by the buckling corrections developed based on MCNP4A is not accurate. Especially, in deriving the effective radial length of fuel stack, the approach appears to be questionable. The use of another code which has cross section libraries with temperature effects seems to be a more straightforward and accurate way of determining the reactivity insertion as a function of moderator and fuel temperatures.

The RAI identifies the potential for an inaccuracy in the DOE method for compensating for the lack of sufficient temperature dependence in the MCNP cross section libraries. However, DOE has mitigated this problem because reactivity tables for the kinetics model in RELAP5 were generated using SAS2H with the SCALE libraries that include Doppler-adjusted cross section evaluations for both uranium and plutonium as well as temperature dependent scattering cross sections for a number of low-mass isotopes. Because SAS2H is a one-dimensional code, buckling corrections are used to represent the three-dimensional effects. The parameters of the buckling corrections are adjusted to better reproduce the k_{eff} of a set of MCNP benchmark cases. For any given configuration of waste package components (intact or degraded) the buckling approximation can be improved by matching more MCNP benchmarks.

DOE believes that the ultimate accuracy of this adjustment methodology is only limited by the fact that it is primarily at one temperature. However, three additional points should be noted in evaluating the ultimate accuracy potential of this methodology.

(a) Some use may be made of the temperature dependence in the ENDF cross section libraries since they are continually being improved by the addition of temperature dependent cross sections for the most neutronically significant elements. With these it will be possible to reflect some temperature dependence in the buckling adjustment process.

(b) The primary sensitivity of RELAP applications is with respect to changes in k_{eff}, not its absolute value. DOE believes that changes in k_{eff} caused by changes in configuration can be represented with reasonable accuracy by changes in MCNP geometry. DOE believes that changes in k_{eff} caused by changes in temperature are best represented by the change in SAS2H geometry.

(c) The modeling of transient criticality is usually acceptable within 50% (RELAP5/MOD3 Code Manual, NUREG/CR-5535) A typical RELAP modeling question is whether some component has been seriously damaged, so there is little need for high precision.

4-45 Clarify whether in the transient criticality analysis method the code biases and uncertainties, in addition to the Monte Carlo uncertainties, are included in all the k_{eff} values.

Examination of CRWMS M&O 1997b and CRWMS M&O 1997e indicates that the change in reactivity might be based on a subcritical initial condition. This is due to subtracting the code bias and uncertainty from k_{eff} = 1for initial condition. This would result in the majority of reactivity being inserted while configuration is in subcritical condition. Starting with critical condition (i.e., 0.95 + 0.05 for bias and uncertainties) and adding the bias and uncertainties to the other transient conditions (e.g., k_{eff} = 1.0189 + 2o + 0.05) would place the reactivity insertion above the critical condition.

Bias and uncertainty are not directly relevant to the transient criticality analysis because the RELAP calculation is only concerned with differences in k_{eff}. DOE believes that bias would not change much with a small change in k_{eff}. The change in k_{eff} for the example reported in Appendix C of the Topical Report was based on changing from a subcritical configuration (uniform oxide distribution) to a critical configuration (settled oxide) with a Δk_{eff} of approximately 0.07. Since that calculation, the design concept has been changed somewhat so that the difference between these configurations is no longer significant, as explained in the answer to RAI 3-19, above. The Topical Report will be revised to clarify this point. Future application of this methodology is expected to continue to follow the principal of computing differences in k_{eff} between possible physical configurations. The parameters for these configurations are expected to be chosen so that they, both before and after the onset of transient criticality, closely straddle the point k_{eff} = 1.

4-46 Discuss the approach for transient criticality analysis for high-enriched spent fuels in view of absence of negative Doppler feedback.

The approach in selecting configuration classes for transient criticality presented in the topical report is with respect to low-enriched commercial spent fuel assemblies. The high-enriched spent fuels, such as DOE-owned spent fuels, will have no negative fuel temperature feedback (i.e., not enough U-238 for Doppler feedback). This type of configuration class must be also analyzed.

The Topical Report is expected to be revised to clarify that the methodology of RELAP5 is expected to be applied to high-enriched SNF. The code is expected to handle a broad range of values for all the feedback coefficients. DOE expects to perform transient evaluations for SNF with upwards of 20% enrichment during FY2000. Although highly enriched fuel does have a small Doppler temperature coefficient, the moderator density reactivity feedback is large.

4-47 Discuss the over-moderation effect within the waste package in view of the large uncertainty associated with the flow rate into the waste package.

Another transient criticality configuration which must be addressed is the over-moderation configuration. Configurations with large flow rate into the waste package and a subsequent seismic event can result in a positive-feedback criticality.

The example of transient criticality in Appendix C of the Topical Report did not directly address the issue of overmoderation. However, in practice the MCNP calculations covers the range of water volumes, starting with the initial configuration and reducing in steps to a value small enough to accommodate the blown-down configuration. Hence, RELAP would automatically calculate any positive void coefficient associated with an overmoderated condition. The Topical Report will be revised to state this.

4-48 Explain how the point neutron kinetics and flow models in RELAP5/MOD3.2 will be adapted for broad applicability to the analysis of internal criticality transients involving all intact and degraded waste forms and packages.

The staff notes that the feedback coefficients in RELAP5/MOD3.2's point neutron kinetics formulation are limited to those feedback mechanisms needed for modeling selected PWR transients. For example, the "void coefficient" formulation assumes that "coolant" and "moderator" are one and the same, which is not valid for certain non-PWR ractor types and likewise not valid for the many intact and degraded waste form/package configurations that involve more than one "moderator/coolant" medium (e.g., see example configurations in Appendix D of the report). The NRC Office of Research, which oversaw the development of RELAP5/MOD32 at INEEL, has noted that significant revisions to the code's neutron kinetics and flow models would be needed for applying the code to other reactor types such as CANDU, RBMK (i.e., Chernobyl), etc. Similar revisions may likewise be needed for applying the code to the full range of criticality transients in the repository. This potential code deficiency is closely related to the previously noted deficiencies in the concept of over/under-moderation which does not address the full range of neutronic phenomena that govern positive and negative feedback effects.

DOE recognizes that the application of RELAP to transient criticality is only appropriate when water is the principal moderator. In addition to the PWR waste package, RELAP5

is intended for application to the BWR waste package, although the accuracy will be reduced because the BWR channels will limit the flow across an assembly that dominates the PWR waste package transient criticality. The dominant flow in the BWR SNF transient criticality will be along the waste package axis inside the assembly channel, with a return flow outside the channel volume to the exit area(s). The turnaround at the ends of the waste package will be modeled conservatively (to produce a component resistance at the high end of the uncertainty range). The resistance of the turnaround and the conservative model of the turnaround are expected to make the peak pressure and temperature considerably higher for a BWR transient criticality than for a PWR SNF transient criticality.

The RELAP application are expected to be limited to configurations with the fuel pins in their nearly intact configuration so that water can flow between the pins. There may be some potential for increasing k_{eff} by precipitation of silica directly on the fuel pins or on basket material. This is expected to be modeled by increasing the resistance of the junction component models used to represent the flow through, and around, the assemblies. When the waste package becomes primarily silica moderated, the coupled thermal-hydraulic-neutronic code mentioned in the responses to RAI's 3-21, 4-49, and 4-53 is expected to be applied instead of RELAP5.

With respect to the coolant issue, it should be noted that in a waste package criticality there is nothing analogous to a coolant, and none is modeled. Inputs to the RELAP waste package criticality runs specify no feedwater or other coolant source.

Section 4.4.2.1 Steady State Criticality

4-49 Discuss the approach for consequences of external criticality, some of which are presented in Probabilistic External Criticality Evaluation report, in the topical report.

The above report presents some qualitative discussion with regard to only an increase in radionuclide inventory. More in-depth quantitative approach is needed to address the steady state external criticality consequence, especially with regard to high enriched spent fuels.

The principal consequence of a steady-state criticality is radionuclide increment. Any large pressure or temperature cannot be sustained on a steady-state basis, and should be evaluated as a transient criticality. As stated in the response to RAI 3-21, for criticality external to the waste package, the Topical Report states that a code with fully coupled thermal, hydraulic and neutronic effects is expected to be used to assess consequences of steady-state criticalities (Section 4.4.1.2). This code is expected to be applicable to any time-dependent criticality, even if it looks more like steady-state than transient.

Section 4.4.2.2 Transient Criticality

4-50 Discuss the approach for addressing consequences of transient criticality such as autocatalytic criticality from possible re-concentration of fissile masses in the near-field and far-field.

Re-concentration of fissile material in the near or far field combined with subsequent sudden flow of water can result in external transient criticality situations. An approach to address possible consequences of this configuration class is needed.

As stated in the responses to RAIs 1-4 and 3-22, DOE expects to evaluate all potentially autocatalytic configurations that can be reached by conceivable scenarios. Preliminary evaluations of external accumulation thus far indicate that such accumulations are not possible for commercial SNF in the near field (CRWMS M&O 1997), and DOE is continuing to explore the possibility of reducing zones in the far-field. Other waste forms are also being evaluated in this respect. CRWMS M&O 1997. *Waste Package Probabilistic Criticality Analysis: Summary Report of Evaluations in 1997.* BBA000000-01717-5705-00015 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19980204.0095.

Section 4.4.3.1 Validation of the Steady-State Criticality Consequence Methodology

4-51 Provide a discussion of the approach used to identify applicable experiments which would quantify the bias and uncertainty associated with the steady-state criticality analysis.

Qualitative discussion with respect to conservatism does not provide the quantitative values for uncertainties and bias which need to be identified. For example, examination of the reference material indicates that there is a large uncertainty associated with predicting the steady state power. The analysis indicated that the power produced can be between 0.5 KW and about 4 KW. The average of these two numbers was used. Other areas of analysis have large uncertainties which need to be quantified and taken into consideration for predicting the consequence of the steady-state criticality.

DOE believes that the bias and uncertainty associated with the calculation of k_{eff} are small in comparison with the other uncertainties in the steady state criticality problem. For the Viability Assessment waste package design that was used for the steady-state criticality consequence calculation in the Topical Report, the principal sources of uncertainty were the following: (1) the particle size in the rockfall surrounding the waste package, (2) the fraction of the waste package outer surface covered by rockfall, (3) the number of assemblies covered by water, (4) the duration of the criticality. The first two of these parameters are the primary determinants of the heat conducted away from the critical waste package. With the new EDA II waste package/EBS design concept, the

rockfall is mitigated by the backfill, which will cover the entire waste package. Once the backfill material is decided the uncertainty associated with parameters 1 and 2 will be removed, but the EDA II design concept is expected to be sensitive to the degree of degradation of the drip shield at the time of the criticality. In keeping with the risk-informed approach, the remaining uncertainties will be quantified as suggested by the RAI and mapped into probability distributions. Because the physical processes are well understood, and the configuration itself is the primary source of uncertainty, DOE believes that no experiments are needed. The bias and uncertainties are planned to be calculated in the validation reports. The Topical Report will be revised to more clearly explain the general process discussed in this response.

Section 4.4.3.2 Validation of the Transient Criticality Consequence Methodology

4-52 Justify the applicability of RELAP5 to the waste package in light of differences in orientation and presence of iron oxide.

As indicated in the above questions, the RELAP5 has been developed for reactor cores with moderator and coolant flow in the direction parallel to fuel assemblies with minimal cross flow across the fuel assemblies. The situation in the waste package is the reverse of that in the core. Applicable experiments need to be identified in order to provide confidence in predicting transient criticality consequences.

As explained in the response to RAI 4-43, any licensing presentations of this sort of analysis will demonstrate accuracy by comparison of the behavior of the component models (that are used to represent cross-flow) with actual test data.

4-53 Provide a discussion of the approach used to identify the super critical experiments which will be used to validate the appropriate transient criticality model.

This section needs to discuss the benchmark experiments which will be used in validating the transient criticality computer code. The discussion should be in terms of area and range of applicability.

The approach used to identify the super-critical experiments, which will be used to validate the appropriate transient criticality model, is based on the benchmarks that the NRC has accepted (RELAP5/MOD3 Code Manual Volume III (Draft), Developmental Assessment Problems, NUREG/CR-5535) . As suggested by previous licensing evaluations using RELAP, if the geometry of the fissile configuration remains in a quasi-static form during the transient, then the spatial and spectral integrals of the neutron kinetics equations are appropriate to model the time dependence of the reactivity transient.

The area and range of applicability of the RELAP5 model includes all reactivity transients. The super-critical experimental condition representing the most severe transient in terms of power is prompt criticality with more than a dollar of reactivity inserted. The benchmarks that have validated the neutron kinetics for prompt criticality are associated with the control rod ejection accident in a reactor. The experimental condition representing the least severe power transient is power escalation in a reactor. By using a very slow escalation, DOE could use some of the sophisticated modeling features (such as the vaporization model) of RELAP to model the steady-state criticality more accurately than was done in Appendix C, Section 5.1 of the Topical Report.

Validation of RELAP will proceed by showing that it exhibits proper behavior over a set of complex examples. DOE is presently using four standard RELAP benchmarks (Marviken III Test 24, Loft Test L3-1 Accumulator Blowdown, SemiScale Natural Circulation Experiment, and MIT Pressurizer Experiment) to secure QA validation for the use of RELAP for transient criticality. Although none of the cases duplicates our conditions, DOE expects to demonstrate that the essential features of the code reproduce experimental results. This process will also identify the uncertainty (and/or bias) in using the code as a predictor.

Ultimately, the validation of a consequence code for repository applications will be concerned with two types of criticality impact: (a) maximum pressure and temperature which can cause damage to the fuel and/or the waste package barriers, thereby enhancing the release of radionuclides, and (b) the maximum energy (or steady-state criticality) to produce the greatest increment in radionuclide inventory. The Topical Report, Sections 4.4.1.1, 4.4.1.2, and Appendix C, Section 5.1, indicate how RELAP5 would be used to estimate impacts of type (a), and how a simpler analysis methodology could be used for the steady state criticality (impacts of type b). As mentioned above, certain features of RELAP5 could enhance the accuracy of the evaluation of type (b) impacts as well.

For external criticality transient consequence evaluation, and other configurations not suited to RELAP analysis, DOE plans to use a coupled thermo-hydraulic-neutronic code. Either the code developed by the University of California Berkeley Nuclear Engineering Department, and mentioned in Section 4.4.1.2 of the Topical Report, or the code developed by Los Alamos National Laboratory (MRKJ, unpublished) may be used. The validation of such a code would include comparison with the standard criticality accident/transient experiments (e.g., Godiva, Kiwi).

DOE plans to revise the Topical Report to reflect the information provided in this response.

Appendix C Example Application of the Methodology for Commercial Spent Nuclear Fuel

C-1 Provide information on plans for geochemical model validation in this example analysis and the Appendix D example analysis (see discussions above on model

validation). This information will make clearer the scope and rigor of the validation approach.

The main body of the Topical Report contains the validation process for the geochemical models. As indicated in the responses to earlier RAIs, DOE will provide the detailed validation information in separate validation reports. The reports referenced in the main body of the Topical Report contain the details of the evaluations performed to date, and some plans for future evaluations. The example in this Appendix is intended only to briefly demonstrate various aspects of the methodology.

The best way to validate the code is to show that it exhibits proper behavior for all the relevant output parameters, over a set of complex examples. It is not critical that the test cases match exactly the time and chemical conditions of the waste package models. It is important that the test cases span the Damköhler number (Da) and Peclet number (Pe) expected in the waste package/near-field models, and the test cases and waste package models involve rate laws and chemical behaviors of similar complexity. In essence, this is a restatement of the principle of similarity. For example, one scenario for precipitation of actinides, in a crushed-tuff invert, involves dissolution of Ca- and Fe(II)-silicates. The Ca may re-precipitate as calcite, indirectly destabilizing actinide carbonate complexes; the Fe(II) may act as a reductant for Pu; and the excess silica may act as a precipitant for U. The dissolution of the silicates is a rate-limiting step, and the density of precipitant is partly controlled by the ratio of the characteristic diffusion or advection time, to the characteristic dissolution time (the Da). In addition, the amount of precipitation is limited by the total availability of Ca and Fe(II) in the system. Thus an appropriate benchmark may be as simple as an experiment that flushes J-13 through crushed Topopah Springs Tuff. The experiment may involve much higher flow rates than are proposed for flow through the invert; however, the same Da can be achieved in the experiment by careful control of the grain size, and thus the surface area (which directly the dissolution rate per gram of reactant).

Section C.1.4.3 Waste Form Degradation Characteristics

C-2 Expand the discussion of the structural and corrosion characteristics of Zircaloy cladding to include the effects of irradiation, reactor water chemistry, operating history, and pre-disposal storage conditions.

Commercial SNF exhibits a wide range of Zircaloy material characteristics, including large variations in the degree of hydriding, oxidation (corrosion), erosion thinning, embrittlement, pallet-cladding interactions, pinhole/crack formation, crud deposition, etc.

As noted in the response to RAI 4-26, future licensing documents relying on cladding credit which will be based on the extensive testing and carefully reviewed modeling programs now ongoing. Clarification will be added to the revision to the Topical Report.

C-3 Clarify the intent of the statement that "At sufficiently high temperatures in an oxidizing environment, the fragments will oxidize ..."

Oxidation of UO_2 does not, in general, require elevated temperatures.

The referenced statement was based upon earlier work, and DOE agrees that it is not valid. It is irrelevant to the topics under discussion in Appendix C. The statement will be deleted from Appendix C.

Section C3.3 Criticality Regression Expression

C-4 Justify the applicability of k_∞ which was based on all isotopes in spent fuel, to spent fuel with the 29 principal isotopes.

The k_∞ regression equation developed by ORNL is based on all the isotopes included in SAS2H. If these results are used for binning the spent fuels with the principal isotope assumption, the k_{eff} appears to be under-predicted.

k_∞ was established as a temporary expedient to distinguish SNF requiring more, or less, criticality control measures. The k_∞ regression was never used for determining probability of criticality. Now that DOE has enough cases of k_{eff} calculation (over 2000 for the various waste forms and various degradation parameters), DOE can quickly estimate k_{eff} for a range of waste forms and a number of ranges for individual degradation parameters by table lookup and interpolation. Therefore, DOE will have no further use for k_∞. The discussion of k_∞ in Appendix C of the Topical Report will be deleted.

C-5 Assess the impact of not including axial burnup profile and the reactor operating history bounding parameter values in the regression equation.

Using single uniform axial burnup profile and nominal values for reactor operating history parameters would result in a regression equation under-predicting the k_{eff} values.

The combined impact of not including axial burnup profiles and bounding depletion parameters in the example evaluation is expected to be, as noted in the RAI, an under-prediction of k_{eff} values. If the example were done including axial profiles and bounding parameters, it would be expected that more potential criticalities would be predicted. The effect of axial burnup profiles on regression equations has been evaluated in an initial evaluation (CRWMS M&O 1998, p. 32, Figure 6.1-1), and the effect was found to be minor. As noted in the response to RAI 3-2, DOE is currently using the multi-node model for commercial SNF criticality calculations for the methodology. The effect of using the bounding depletion parameters has not been assessed yet, since the bound parameters have not been established for any commercial SNF.

DOE plans to update the example evaluations in the revision to the Topical Report to incorporate the available axial burnup profile and bounding depletion parameter information, or note what the expected effect of not including the information would be.

CRWMS M&O 1998. *Supplemental Criticality Evaluation for Degraded Internal Configurations of a 21 PWR Waste Package*. BBA000000-01717-0210-00022 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19980918.0086.

Section C4.1 Probability Estimation

C-6 Evaluate the potential for axial displacement of the disposal control rods relative to the active fuel.

The topical report takes credit for the control rods as the basis for not considering the most reactive fuel, i.e., fuel with burnup below the loading curve, as part of the population of PWR fuel capable of exceeding the critical limit. Any upset or degradation mechanisms that could produce axial displacement (e.g., tilting of the package or basket) should be identified.

If control rods are used, one or more mechanisms are expected to be used to ensure that they cannot be axially displaced while the fuel is intact. Such mechanisms may include friction fitting of the control rods, pinning at one end, or space limitaiton inside the waste package. Furthermore, based on current preliminary designs, there is not expected to be enough room in the fuel assembly cavity of a waste package for the rods to slide out. Nor have any credible degraded waste package scenarios, that could result in such a displacement, been identifed. The criticality evaluations used to support the License Application for waste package with disposal control rods will need to fully evaluate this possibility.

DOE plans to update the example evaluations in the revision to the Topical Report to incorporate any available credible degraded waste package scenarios which would result in axial or radial displacement of disposal control rods.

Section C5.1 Criticality Consequence Estimation

C-7 Justify limiting the reactivity insertion scenario to the relatively slow ones described here.

The reasoning for not considering rapid reactivity insertions resulting from sudden movements such as those caused by collapse of the degrading basket structures, rock fails, etc., should be provided.

Justifications for the reactivity insertion rates used here have been discussed in the responses to RAIs 3-19, 3-20, and 4-42.

C-8 Justify the apparent conclusion that long-term steady-state criticalities bound the consequences of all criticality events.

During a long-term steady-state criticality, many of the radionuclides produced will decay or be burned out. High power transient events resulting from rapid reactivity insertions and/or autocatalytic effects, perhaps in conjunction with steady-state criticalities, have a potential to produce a burst of short-lived fission products and actinides (i.e., short-half life equates to high activity) as well as transient thermal-mechanical effects that may promote their early release from the repository.

The additional burnup from the long-term steady-state criticality was approximately 820 MWd/mtU, while the additional burnup from the transient criticality was 1.8×10^{-3} MWd/mtU (CRWMS M&O 1996, CRWMS M&O 1998a). As can be seen in the response to RAI C-9, higher amounts of the short-lived radionuclides are present at the end of the steady-state example than at the end of the transient example. This is also true for the long-lived radionuclides important to performance assessment.

The possibility of the criticality event accelerating the release of the additional radionuclides generated (e.g., by damaging previously intact fuel cladding) has already essentially been addressed in the TSPA evaluation reported in Appendix C. This evaluation (CRWMS M&O 1998b) assumed that the additional radionuclides are available for removal from the waste package immediately following termination of the criticality.

CRWMS M&O 1996. *Second Waste Package Probabilistic Criticality Analysis: Generation and Evaluation of Internal Criticality Configurations.* BBA000000-01717-2200-00005 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19960924.0193.

CRWMS M&O 1998a. *Criticality Consequence Analysis Involving Intact PWR SNF in a Degraded 21 PWR Assembly Waste Package.* BBA000000-01717-0200-00057 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19980106.0331.

CRWMS M&O 1998b. *TSPA-VA Total System Model Base Case Modified to Include Nuclear Criticality Disruption of the Repository.* Las Vegas, Nevada: CRWMS M&O. DTN: SNT05072098001.004. ACC: MOL.19981125.0027

C-9 Verify that short-lived isotopes arising in high-power transients are considered in evaluating dose consequences.

It is not clear that the short-lived isotopes potentially important to the dose consequences of rapid transient events (e.g., Kr-85, I-131, Sr-89, Cs-134) are included in the isotopes evaluated under this methodology. This section limits its evaluation to the 36 TSPA-95 isotopes.

Radionuclide inventories for the short-lived radionuclides were also calculated as part of the consequence evaluations performed for the steady-state and transient criticality examples. The table below provides information on the inventories of various short-lived radionuclides at the termination of each of the example criticality events shown in Appendix C. As is indicated on Page C-57 of Appendix C, gas travel times from the repository block to the surface were estimated to be in the range of 200 to 600 years. This is the same order of magnitude as the 300 year minimum groundwater travel time to the accessible environment given in (DOE 1998, Figure 4-18). Therefore, both gas and groundwater travel times are at least an order of magnitude greater than the half-life of any of the radionuclides listed below, it is not expected that any significant quantity of these radionuclides could reach the surface via a gaseous pathway.

Radionuclide	Half-Life (GE 1989)	Ci/assembly at end of criticality event	
		Steady-state example (CRWMS M&O 1996, Att. X)	Transient example (CRWMS M&O 1997, long.out)
H-3	12.3 y	1.1×10^{-2}	1.2×10^{-5}
Kr-85	10.7 y	0.19	1.1×10^{-5}
I-131	8.0 d	2.73	-
Sr-89	50.5 d	3.29	6.7×10^{-2}
Sr-90	29.1 y	3.97	1.8×10^{-3}
Cs-134	2.1 y	1.15	1.4×10^{-6}
Cs-137	30.2 y	5.49	2.0×10^{-3}
Ru-106	1.0 y	1.42	1.7×10^{-2}

Furthermore, even if all of the fuel rods in a 21 PWR waste package were assumed to fail immediately after termination of the criticality, and all of the radionuclides were assumed to be quickly transported to the surface, the dose from the event would still be very low. The total effective dose at a distance of 5 km from such a non-mechanistic scenario would be < 1 mrem for the steady-state example, and < 10^{-4} mrem for the transient example. The total effective doses were estimated by summing the inhalation and submersion doses for the radionuclides of interest. The inhalation dose from isotope i (ID_i) is estimated using:

$$ID_i = S_i * N * IDCF_i * RF_i * BR * [\chi/Q]_{5km}$$

The submersion dose from isotope i (SD_i) is estimated using:

$$SD_i = S_i * N * SDCF_i * RF_i * [\chi/Q]_{5km}$$

The parameters in the above two equations are as follows:

S_i = Inventory of radionuclide i per assembly (Ci)
N = Number of PWR assemblies in waste package (21)

$IDCF_i$ = Effective inhalation dose conversion factor for isotope i (From EPA 1988, pp. 122-137; for isotopes with multiple lung clearance classes, the class with the highest effective DCF was used).

Nuclide	Effective IDCF (Sv/Bq)
H-3	1.73E-11
Kr-85	N/a
I-131	8.89E-9
Sr-89	1.12E-8
Sr-90	3.51E-7
Cs-134	1.25E-8
Cs-137	8.63E-9
Ru-106	1.29E-7

Conversion to mrem/μCi made by multiplying by 3.7E9.

$SDCF_y$ = Effective submersion dose conversion factor for isotope i (From EPA 1993, pp. -65)

Nuclide	Effective SDCF (Sv per Bq*sec*m^{-3})
H-3	3.31E-19
Kr-85	1.19E-16
I-131	1.82E-14
Sr-89	7.73E-17
Sr-90	7.53E-18
Cs-134	7.57E-14
Cs-137	7.74E-18
Ru-106	N/a

Conversion to mrem per μCi*y*cm^{-3} made by multiplying by 1.168E23

RF_i = Release fraction for isotope i, defined as the fraction of the inventory of the radionuclide that is present in the waste form that is released to the environment during an event (see Table 7.1, NUREG 1536 or CRWMS M&O 1998d, p. 8)

Nuclide	Release Fraction
H-3	0.3
Kr-85	0.3
Iodines	0.1
Cs and Sr	2.3E-5
Ru	1.5E-5

BR	=	Breathing rate; 3.3E-4 m^3/sec (CRWMS M&O 1998d, p.9)
$[\chi/Q]_{5km}$	=	Best-estimate atmospheric dispersion factor at 5 km distance for assumed meteorological conditions and duration of release; 1.44E-5 sec/m^3 (CRWMS M&O 1998d, p. 9)

Additional credit for dispersion in the fracture network as the gas is transported from the drift to the surface, and "filtration" by particle filled or dead-end fractures, would further reduce these estimates.

CRWMS M&O 1996. *SAS2H Generated Isotopic Concentrations for B&W 15x15 PWR Assembly.* BBA000000-01717-0200-00012 REV 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19961218.0190.

CRWMS M&O 1997. *Criticality Consequence Analysis Involving Intact PWR SNF in a Degraded 21 PWR Assembly Waste Package.* BBA000000-01717-0200-00057 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19980106.0331.

CRWMS M&O 1998b. *Preliminary Preclosure Design Basis Event Calculations for the Monitored Geologic Repository.* BC0000000-01717-0210-00001 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19981002.0001.

EPA 1988. *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion: Federal Guidance Report No. 11.* EPA-520/1-88-020. Washington, DC: EPA. TIC: 203350.

EPA 1993. *External Exposure to Radionuclides in Air, Water, and Soil: Federal Guidance Report No. 12.* EPA 402-R-93-081. Washington, DC: EPA. TIC: 225472.

DOE (1998), *Viability Assessment of a Repository at Yucca Mountain, Volume 3, Total System Performance Assessment,* DOE/RW-0508/V3.

C-10 Justify why the presence of boron and corrosion products dissolved or suspended in the water and that can affect the moderator void coefficient of reactivity is not addressed in the TR.

The presence of absorbers in the "moderator" can produce a positive void reactivity, resulting in autocatalytic feedback effects that are apparently not considered in the proposed methodology.

The proposed methodology can consider the potential for a positive void coefficient provided by dissolved neutron absorber. The nominal tool for evaluating the consequences of a transient internal criticality, RELAP 5, is capable of exhibiting behavior reflecting the presence of dissolved neutron absorber (e.g., boron) in moderating water. This capability was not used in the transient criticality example of Appendix C of the Topical Report, because the waste package is only exposed to such a criticality threat

for a relatively brief period of its history. Most of the time that there is boron in the waste package, the boron will serve to prevent criticality entirely. When the boron has essentially been completely removed from the water, and from the waste package, there may be a criticality, but there can no longer be a positive void coefficient. In the brief period between these conditions there could be such a positive void coefficient, and this possibility is expected to be evaluated in the documents that will support the License Application.

C-11 Justify the assumption of one-year decay time in assessing the consequences of a fast transient criticality. In particular, explain how the one-year decay time bounds the travel times of all important radionuclides.

Fast transient criticality events resulting from rapid reactivity insertions and/or autocatalytic effects have a potential to produce a burst of short-lived fission products and actinides. However, it is not clear why the one-year decay time assumption is used in assessing the consequences in this case.

The assumption of one-year decay time was considered conservative because several of the long-lived radionuclides shown are not at their peak activities at the time the criticality event is terminated, but reach their peak at various times during the first year (e.g., Pu-238 peaks between 0 and 60 days as a result of build-in from the decay of Np-238 and Cm-242).

C-12 Explain the design modification needed in light of the change in the radionuclide inventory indicated by Table C-16.

The criticality consequence design criterion listed in Section 1.2 states that "the expected radionuclide increase from any criticality event will be less than 10 percent..." Table C-16 indicates a net increase of more than 18% for the five isotopes which are important to the repository performance. Given the results, verify that a design change is needed.

The criticality consequence criterion discussed in Sections 1.2 and 3.6.3 requires that the increase in radionuclide inventory be compared with the total inventory for that waste form that is available for release in the entire repository, not in a single package. Furthermore, as is shown in Figure 1-1, failure to meet this criterion requires that the increment in radionuclide inventory be considered in the TSPA. The purpose of this criterion is to avoid performing unnecessary TSPA analyses if the radionuclide increment resulting from the criticality is extremely small.

Section C6.1 Total System Performance Assessment Dose Estimation

C-13 Evaluate travel times for gases to the surface for the cases of worst-case disruption of the EBS and repository environment from energetic criticality transients.

As stated in the discussion, "gaseous fission products such as 85Kr are ignored because only a small amount are produced..." However, there are others that will be produced in significant amounts. Therefore, an evaluation of the travel times for these gaseous fission products to the surface becomes important.

Please see the response to RAI C-9.

C-14 Verify that Figures C-34 and C-35 indicate the incremental dose from just increasing the radionuclide inventory by the numbers in Table C-16 in one waste package which is the result of a single criticality. In addition, discuss the risk from multiple waste packages becoming critical because of juvenile failures.

Figures C-34 and C-35 represent the dose history that would be expected if only the additional radionuclides generated during the 10,000-year criticality event were immediately released from the waste package at the time that the criticality event terminated (25,000 years). Other radionuclides, in addition to those shown in Table C-16, were also considered in the TSPA evaluation. A complete list of the radionuclides considered, and the curie inventory for each, is given in the table below. However, the first four radionuclides shown in Table C-16 were the dominant contributors to additional dose from the criticality. The Pu-242 difference was included in Table C-16 to demonstrate that the criticality event actually produced a reduction in some of the radionuclides that were considered important to TSPA.

Isotope	Curie Inventory at 25,000 years			
	Single Assembly* After 10,000 Year Criticality	Single Assembly* Decay Only	Single Assembly* Increase From Criticality	21 PWR WP Increase From Criticality
Pa-231	1.40E-02	6.30E-03	7.70E-03	1.62E-01
C-14	2.40E-06	1.60E-06	8.00E-07	1.68E-05
U-234	7.20E-01	6.50E-01	7.00E-02	1.47E+00
Pu-239	8.70E+01	8.00E+01	7.00E+00	1.47E+02
Se-79	1.20E-01	1.10E-01	1.00E-02	2.10E-01
Tc-99	3.80E+00	3.60E+00	2.00E-01	4.20E+00
I-129	9.20E-03	8.80E-03	4.00E-04	8.40E-03
Np-237	3.90E-01	3.80E-01	1.00E-02	2.10E-01

* 3 wt% U-235 enrichment, 20 GWd/mtu initial burnup

Juvenile/early waste package failures were not considered in the example evaluation included in Appendix C because a final rate had not yet been developed at the time the criticality probability estimate was performed. However, since the methodology requires that the waste package failure distribution used in the criticality probability calculations be consistent with that used in the TSPA, this issue is expected to be addressed in the references to the License Application. Furthermore, with the change in the waste package barrier materials, it is expected that early waste package failure will be the

dominant contributor to the criticality probability during the first 10,000 years. However, this probability is still expected to be very small for the following reasons:

a) Only a small fraction of packages (on the order of 10^{-4} per waste package) would be expected to have a manufacturing or handling induced defect that could lead to early failure;
b) The defective package must be located under a dripping fracture to be of concern for criticality;
c) The defect must result in a failure that is capable of passing a significant fraction of the water dripping on the waste package into the interior, and must be located such that ponding of water within the waste package is possible;
d) The borated stainless steel must become sufficiently degraded and the boron removed from the waste package;
e) The waste package must remain flooded long enough for the borated stainless steel to degrade to the point where criticality is possible;
f) The defective waste package must contain fuel that is capable of exceeding the critical limit if the package is flooded, the borated stainless steel is sufficiently degraded, and the boron has been removed from the waste package.

A rough estimate can be made using the probabilities given in Table 4-4 of the Topical Report, and substituting the one juvenile failure at 1,000 years assumed in the TSPA for P_{breach}. Using this crude approximation, the contribution to cumulative probability of criticality at 10,000 years from juvenile failure for the no-loading curve case would be $\sim 2 \times 10^{-7}$ per PWR waste package.

Section C7.0 Conclusion

C-15 The example and the topical report does not address the classes of criticality events with potentially high consequences.

In particular, the report does not give adequate attention to sudden reactivity insertions and the full range of mechanisms for positive reactivity feedback (i.e., autocatalytic criticality).

The technique used to identify and evaluate all rapid insertion rate mechanisms is discussed in the responses to RAIs 3-19, 3-20, and 4-42.

Appendix D Example Application of the Methodology for DOE Spent Nuclear Fuel
Section D2.2 External Configurations

D-1 Correct the reference to the discussion in Appendix C of external configurations. Neither Appendices C or D evaluate external criticality.

In Section 2.2, it is stated that eventual discussions of external criticality *will include configurations similar to those discussed in Appendix C*, Sections 2.2.2 and 2.2.3 of Appendix C discuss external configurations with the potential for criticality. There is also reference to the documents in which the criticality evaluations associated with these configurations are described.

Section D3.1 Evaluation of Critical Configurations

D-2 Clarify the application of the methodology in this example, with respect to the flow chart presented in Figure 1.1 in the main body if the report, page 1-10.

It appears, in this example, that design changes were made in the choice of poison material ($GdPO_4$) and its concentration, without having evaluated configurations with respect to the probability criterion. This suggests a departure from the methodology described in the flow chart. The staff notes that the flow chart may need to be revised to reflect the fact that the need for design changes may become apparent much earlier in the process.

The example application did not follow the full path of the disposal criticality methodology as presented in Figure 1-1. The evaluation included a conservative assumption, based on engineering judgement, that the probability criterion would not be satisfied, and implemented a design option for the DOE SNF Canister. So there was not really a departure from the methodology, only a conservative simplification. It is likely that the same results would have been reached if the probability evaluation had been performed.

DOE does not plan to modify the flow chart to specifically allow for conservative assumptions based upon engineering judgement. The flow chart of the methodology is sufficient (has the steps to help ensure the health and safety of the public) without the added optional steps.

Attachment A

Commercial SNF Internal Configuration Validation Planned Workscope

I. Isotopic Validation

A. Code-to-Code Comparison

Prior to validating the isotopic depletion model with chemical assay data, the SAS2H one-dimensional depletion code is expected to be compared to a more detailed two-dimensional isotopic depletion code. The objective of this comparison will be to evaluate the analytical assumptions and approximations when applying a one-dimensional depletion code to model heterogeneous fuel types with burnable thermal neutron absorbers. In particular, the impact of; lattice smearing, gadolinia fuel depletion, moderator density, control blade insertion during depletion and history effects are expected to be reviewed. Code-to-code comparisons of isotopic concentrations are expected to be performed to demonstrate and quantify depletion endpoint agreement. Isotopic concentration values are expected to be used as input to a waste package criticality model to compare integral k_{eff} values resulting from the various modeling assumptions and approximations.

Information from this section is expected to result in an improved understanding of the effects of the analytical assumptions and approximations used in setting up input decks for analyzing the diverse chemical assays on a consistent basis.

B. Chemical Assays

Fifty-four chemical assay samples were obtained from irradiated fuel assemblies discharged from seven different Pressurized Water Reactors (PWRs) (Reference 2). Thirty chemical assay samples were obtained from irradiated fuel assemblies discharged from three different Boiling Water Reactors (BWRs) (Reference 3). Additional chemical assay samples are expected to be analyzed. These additional assays include 13 PWR samples with enrichments of 4.00 wt% U-235 and 4.65 wt% U-235 and rod average burnups ranging from 27.0 GWd/mtU to 48.5 GWd/mtU. The additional assays also include 13 BWR samples with uranium enrichments of 3.00 wt% U-235 and 3.80 wt% U-235, rod average burnups of about 62 GWd/mtU and about 70 GWd/mtU, and one sample of 2.00 wt% gadolinia. The relevant characteristics and burnup histories of these samples are expected to be modeled in SAS2H and a two-dimensional depletion code.

C. Statistical Analysis

Statistical analyses are expected to be performed on the isotopic validation benchmark set. Statistical tests for outliers and normality are expected to be performed to identify anomalous data points or non-normal data sets.

The measured and calculated isotopic results are expected to be used as input to a consistent waste package criticality model to compare integral k_{eff} effects resulting from the various modeling assumptions and approximations and be used to determine a Δk bias.

D. Isotopic Distribution Effects

Quantification of isotopic distribution effects on criticality calculations are expected to be performed for various isotopic models. Variations of spectrum, exposure history, and modeling approaches are expected to be examined for impact on isotopic distribution. Variations in isotopic distributions for actinides and fission products are expected to be examined for impact on prediction of k_{eff}. Together with other isotopic validation activities, these analyses are expected to address the compensating effects issue regarding the integral criticality calculation method.

E. Decay and Branching Fraction Uncertainty

The effects of uncertainties in the half-life and branching fractions used in predicting postclosure isotopic concentrations are expected to be evaluated by a statistical method (using Monte Carlo). The approach used is expected to model the entire system of isotopic decay with all of the parent-daughter relationships. Together with the nominal isotopic values, a Δk bias can be determined to account for these uncertainties.

References:

1) ANSI/ANS-8.1-1998, Nuclear Criticality Safety in Operations with Fissionable Material Outside Reactors.
2) DOE/RW-0497, Isotopic and Criticality Validation for PWR Actinide-Only Burnup Credit, May 1997.
3) ORNL/TM-13315, Validation of SCALE (SAS2H) Isotopic Predictions for BWR Spent Fuel, September 1998.

II. Criticality Validation

A. Criticality Benchmarks

Waste form-specific benchmark subsets are expected to be defined (from the total Benchmark Database, which consists of LCEs, PWR CRCs, and BWR CRCs) for each applicable scenario/waste class from the master scenario list. The subset development is expected to consider such aspects as material type, geometry and neutron spectrum.

Example:	PWR Intact Fuel Lattice:	PWR CRCs, LEU rod lattice LCEs, MOX rod lattice LCEs
	PWR Fully Degraded Fuel:	LEU homogenous LCEs and MOX

Solutions homogenous LCEs

1. Benchmark Descriptions
   The benchmarks within the given subsets are expected to be characterized based on various parameters (e.g., flux spectrum, fission spectrum, material type and geometry).
2. Adequacy of Benchmarks
   The benchmarks within the given subsets are expected to be compared to the expected configurations. These comparisons are expected to consider a variety of parameters such as isotopic concentrations, actinide ratios (to U-235) and neutron spectrum.

B. Statistical Analyses

Waste form/waste class-specific benchmark subsets are expected to be analyzed for the purposes of identifying trends in the data.

1. Equality Analysis
   The benchmarks within the given subsets are expected to be analyzed to evaluate the possibility of combining subsets. Potential combinations include, but may not be limited to, combining like geometry MOX and LEU LCE subsets.
2. Trending Analyses
   The subsets are expected to be analyzed for trends that may exist within the subsets. Parameters to be considered include spectral parameters (e.g. absorption, leakage and fission spectrums and neutron spectrum ratios), material type parameters (e.g. enrichment, burnup, actinide ratios, plutonium concentrations, and boron concentrations) and geometry parameters (e.g. fuel pellet diameter, fuel rod pitch, and fuel rod pitch to fuel pellet diameter ratios).

C. Development of Critical Limits

Critical limits are expected to be developed using the results of the trending analyses discussed in Section II.B.2 of this attachment, and considering a variety of potential sources of uncertainty (e.g., half-life/branching fraction decay uncertainties and temperature uncertainties). The statistical analyses are expected to be used to determine the combined bias of the computer code and cross section set. This is expected to result in a Δk_{bias}. The uncertainties are expected to be documented in the form of Δk_{eff}. These adjustments are expected to result in a final critical limit ($CL = 1.0 - \Delta k_{bias} - \Delta k_{eff}$).

D. Range of Applicability

A systematic approach is expected to be used to identify the Range of Applicability for every waste form/waste class pairing. This is expected to involve considering trended and un-trended parameters. For trended parameters, both the benchmarked range and the trended range are expected to be considered.

The subset characterizations and the statistical analyses are expected to define

the Range of Applicability for the calculated critical limits. The Range of Applicability is expected to include both trended and non-trended parameters. The parameters may include, but are not limited to: isotopic composition, burnup, initial enrichment (wt% ^{235}U), cooling time, fuel temperature, fuel type, cladding type, fuel density, geometry type (e.g. square lattice, triangular lattice, homogeneous solution), assembly and/or fuel rod pitch, fuel pellet diameter, fuel rod pitch to fuel pellet diameter ratio, absorber types, and absorber concentrations.

A procedure for extending the range of applicability is expected to be provided. The extension process may include a code-to-code comparison in addition to physical explanations for the trend and statistical analyses of the trend. The procedure is expected to also include a method for determining the penalty ($\Delta k_{penalty}$) to be included in the extended range.

III. Application Model

The final step in the criticality validation is expected to be to define a application model and demonstrate that the application model is bounding. This is expected to involve analyzing the CRCs and a number of scenarios in the waste package from the initial enrichment, through the burnup and decay calculations, and demonstrating the reactivity of the CRCs and the waste package are not underestimated for any of the analyzed scenarios.

IV. Validation Reports

The results of the work described in Sections I and II of this attachment are expected to be documented in validation reports, one for BWRs and one for PWRs. The reports are expected to include sections or volumes specific to the validation of the isotopic and criticality models.

Note: This outline of planned workscope is for Commercial Spent Nuclear Fuel (SNF) Internal Configuration only. It is offered as an example. DOE recognizes that additional validation reports are expected to be necessary. These may include validation reports for: DOE Research Reactor SNF Internal Configuration, DOE Plutonium High Level Waste (HLW) Internal Configuration, and Degraded SNF/HLW External Configuration. There are no plans to have a validation report for Naval SNF, separate from the *Addendum to the Disposal Criticality Analysis Methodology Topical Report for Naval Spent Nuclear Fuel.*