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

I. OBJECTIVE

This chapter provides guidance for the author(s) of Chapter 6 of the License Application. The License Application will be submitted to the U.S. Nuclear Regulatory Commission (NRC) for authorization to construct a geologic high-level radioactive waste repository at Yucca Mountain.

This chapter discusses the engineered barrier system excluding the waste packages, which are discussed in Chapter 5 of the License Application. The author(s) will need to demonstrate that the U.S. Department of Energy (DOE) has detailed designs for the engineered barrier system (excluding the waste package) and design analyses sufficient to provide reasonable assurance of the adequacy of the design in supporting repository performance objectives and meeting regulatory design requirements.

Because an engineered barrier system designed to perform for thousands of years has no NRC licensing precedent, the required level of detail of information to be presented and referenced in Chapter 6 of the License Application is expected to be substantial, as discussed in more detail in the following sections.

II. GUIDANCE TOPICS

The following topics will be discussed in Chapter 6 of the License Application:

- 6.0 PURPOSE, SUMMARY, AND CONCLUSIONS
- 6.1 ENGINEERED BARRIER SYSTEM ENVIRONMENT SUMMARY DESCRIPTION
- 6.2 ENGINEERED BARRIER SYSTEM (EXCLUDING WASTE PACKAGE) DESIGN
  - 6.2.1 Design Bases
  - 6.2.2 Design Description
  - 6.2.3 Design Evaluation

III. REGULATORY REQUIREMENTS

*****IMPORTANT NOTE *****

At the time of development of this document, it appears likely that NRC will issue a new, site-specific regulation for Yucca Mountain in 1999. At the time the new 10 CFR 63 is issued, 10 CFR 60 will be modified to state that it is not applicable to Yucca Mountain. Details about the new regulation are not known. However, it has been determined prudent to develop the working draft License Application in accordance with the best available estimate of the form of the new

regulation. To avoid confusing references and to allow production of the technical guidance document in time to support training for the working draft License Application, the text of the technical guidance document that follows this note has not been revised to reflect the postulated form of the new regulation. Instead, this note provides guidance needed by working draft License Application authors to comply with management direction on addressing the new regulation. The paragraphs that follow explain changes in the regulation with respect to the existing 10 CFR 60 that affect this technical guidance document chapter and provide guidance on how to address the changes in the working draft License Application. Where guidance in this note conflicts with guidance in the remainder of this chapter of the technical guidance document, the guidance in the note is to govern. Any questions should be addressed to Management and Operating Contractor (M&O) Licensing.

Changes to Regulations Applicable to Chapter 6

The following 10 CFR 60 requirements applicable to Chapter 6 and listed below in Section III are assumed to not be carried over to the new regulation: 60.113, 60.130, 60.131, 60.133, and 60.135.

In place of these specific requirements, the new regulation is expected to require a discussion of the contribution of the individual barriers to the overall performance of the repository. The manner in which that contribution is to be described has not been specified, though an NRC review plan on the subject may be issued sometime in the future.

Guidance to Working Draft License Application Authors for Addressing Changes

- Disregard guidance in Section IV regarding demonstrating compliance with any of the regulations assumed to be deleted as described above. Instead, discuss the performance of the engineered barrier system and its components in supporting overall repository performance.

- Address the technical issue of each of the regulations in Section III, including those expected to be deleted, but do not relate the discussion to demonstrating compliance with any of the deleted regulations. Continue to explicitly address the regulations in Section III that are not expected to be deleted.

- Detailed discussion of design criteria, design bases, and performance is still required as per the existing guidance in Section IV. Discussion should still include detailed discussions of the phenomena that affect that performance.

- No sections of Chapter 6 are expected to be deleted as a result of this note.

***** END OF NOTE *****

Sections of the regulations that are to be addressed fully or in part in Chapter 6 of the License Application are reproduced below. These are verbatim extracts from the regulations. Paragraph numbers refer to Title 10 of the U.S. Code of Federal Regulations (CFR). In addition, Appendix A of this technical guidance document provides a cross-reference between the regulations in 10

CFR 60 and the specific License Application sections that address these regulations. Since revisions of regulations are not uncommon, preparers of the License Application should verify that current versions are used.

10 CFR Part 60, Disposal of High-Level Radioactive Wastes in Geologic Repositories
60.2 Definitions.

Accessible environment means:

(1) The atmosphere;

(2) The land surface;

(3) Surface water;

(4) Oceans; and

(5) The portion of the lithosphere that is outside the postclosure controlled area.

Barrier means any material or structure that prevents or substantially delays movement of water or radionuclides.

Engineered barrier system means the waste packages and the underground facility.

Underground facility means the underground structure, including openings and backfill materials, but excluding shafts, boreholes, and their seals.

Subpart B - Licenses
License Applications
60.21 - Content of application.

(c) The Safety Analysis Report shall include:

(1) A description and assessment of the site at which the proposed geologic repository operations area is to be located with appropriate attention to those features of the site that might affect geologic repository operations area design and performance. The description of the site shall identify the location of the geologic repository operations area with respect to the boundary of the accessible environment.

(ii) The assessment shall contain...

(C) An evaluation of the performance of the proposed geologic repository for the period after permanent closure, assuming anticipated processes and events, giving the rates and quantities of releases of radionuclides to the accessible environment as a function of time; and a similar evaluation which assumes the occurrence of unanticipated processes and events.

(D) The effectiveness of engineered and natural barriers, including barriers that may not be themselves a part of the geologic repository operations area, against the release of radioactive material to the environment. The analysis shall also include a comparative evaluation of alternatives to the major design features that are important to waste isolation, with particular attention to the

alternatives that would provide longer radionuclide containment and isolation.

(E) An analysis of the performance of the major design structures, systems, and components, both surface and subsurface, to identify those that are important to safety. For the purposes of this analysis, it shall be assumed that operations at the geologic repository operations area will be carried out at the maximum capacity and rate of receipt of radioactive waste stated in the application.

(F) An explanation of measures used to support the models used to perform the assessments required in paragraphs (A) through (D). Analyses and models that will be used to predict future conditions and changes in the geologic setting shall be supported by using an appropriate combination of such methods as field tests, in situ tests, laboratory tests which are representative of field conditions, monitoring data, and natural analog studies.

(2) A description and discussion of the design, both surface and subsurface, of the geologic repository operations area including:

(i) the principal design criteria and their relationship to any general performance objectives promulgated by the Commission,

(ii) the design bases and the relation of the design bases to the principal design criteria,

(iii) information relative to materials of construction (including geologic media, general arrangements, and approximate dimensions), and

(iv) codes and standards that DOE proposes to apply to the design and construction of the geologic repository operations area.

(3) A description and analysis of the design and performance requirements for structures, systems, and components of the geologic repository that are important to safety...

(4) A description of the quality assurance program to be applied to the structures, systems, and components important to safety and to the engineered and natural barriers important to waste isolation.

(6) An identification and justification for the selection of those variables, conditions, or other items which are determined to be probable subjects of license specifications. Special attention shall be given to those items that may significantly influence the final design.

(12) A description of plans for retrieval and alternate storage of the radioactive wastes should the geologic repository prove to be unsuitable for disposal of radioactive waste.

(14) An identification of those structures, systems, and components of the geologic repository, both surface and subsurface, which require research and development to confirm the

adequacy of design. For structures, systems, and components important to safety and for the engineered and natural barriers important to waste isolation, the DOE shall provide a detailed description of the programs designed to resolve safety questions, including a schedule indicating when these questions would be resolved.

Construction Authorization

60.31 Construction Authorization.

Upon review and consideration of an application and environmental impact statement submitted under this part, the Commission may authorize construction if it determines:

(a) *Safety.* That there is reasonable assurance that the types and amounts of radioactive materials described in the application can be received, possessed, and disposed of in a geologic repository operations area of the design proposed without unreasonable risk to the health and safety of the public. In arriving at this determination, the Commission shall consider whether:

(1) the DOE has described the proposed geologic repository including but not limited to:...(iii) the principal architectural and engineering criteria for the design of the geologic repository operations area; (v) features or components incorporated in the design for the protection of the health and safety of the public.

(2) The site and design comply with the performance objectives and criteria contained in Subpart E of this part.

Subpart E - Technical Criteria Design Criteria for the Geologic Repository Operations Area

60.111 Performance of the geologic repository operations area through permanent closure.

(a) Protection against radiation exposures and releases of radioactive material. The geologic repository operations area shall be designed so that until permanent closure has been completed, radiation exposures and radiation levels, and releases of radioactive materials to unrestricted areas, will be maintained within the limits specified in part 20 of this chapter and such generally applicable environmental standards for radioactivity as may have been established by Environmental Protection Agency.

(b) Retrievability of waste. (1) The geologic repository operations area shall be designed to preserve the option of waste retrieval throughout the period during which wastes are being emplaced and, thereafter, until the completion of a performance confirmation program and Commission review of the information obtained from such a program. To satisfy this objective, the geologic repository operations area shall be designed so that any or all of the emplaced waste could be retrieved on a reasonable

schedule starting at any time up to 50 years after waste emplacement operations are initiated, unless a different time period is approved or specified by the Commission. This different time period may be established on a case-by-case basis consistent with the emplacement schedule and the planned performance confirmation program.

(2) This requirement shall not preclude decisions by the Commission to allow backfilling part or all of, or permanent closure of, the geologic repository operations area prior to the end of the period of design for retrievability.

(3) For purposes of this paragraph, a reasonable schedule for retrieval is one that would permit retrieval in about the same time as that devoted to construction of the geologic repository operations area and the emplacement of wastes.

60.112 Overall system performance objective for the geologic repository after permanent closure.

The geologic setting shall be selected and the engineered barrier system and the shafts, boreholes and their seals shall be designed to assure that releases of radioactive materials to the accessible environment following permanent closure conform to such generally applicable environmental standards for radioactivity as may have been established by the Environmental Protection Agency with respect to both anticipated processes and events and unanticipated processes and events.

60.113 Performance of particular barriers after permanent closure.

(a) General provisions. (1) Engineered barrier system. (i) The engineered barrier system shall be designed so that assuming anticipated processes and events: (A) Containment of HLW will be substantially complete during the period when radiation and thermal conditions in the engineered barrier system are dominated by fission product decay; and (B) any release of radionuclides from the engineered barrier system shall be a gradual process which results in small fractional releases to the geologic setting over long times.

(ii) In satisfying the preceding requirement, the engineered barrier system shall be designed, assuming anticipated processes and events, so that:

(A) Containment of HLW within the waste packages will be substantially complete for a period to be determined by the Commission taking into account the factors specified in §60.113(b) provided, that such period shall be not less than 300 years nor more than 1,000 years after permanent closure of the geologic repository; and

(B) The release rate of any radionuclide from the engineered barrier system following the containment period shall not exceed one part in 100,000 per year of the inventory of that radionuclide calculated to be present at 1,000 years following permanent closure, or such other fraction of the inventory as may be approved or specified by the Commission; provided, that this requirement does not apply to any radionuclide which is released at a rate less than 0.1% of the calculated total release rate limit. The calculated total release rate limit shall be taken to be one part in 100,000 per year of the inventory of radioactive waste, originally emplaced in the underground facility, that remains after 1,000 years of radioactive decay.

(2) Geologic setting. The geologic repository shall be located so that pre-waste-emplacement groundwater travel time along the fastest path of likely radionuclide travel from the disturbed zone to the accessible environment shall be at least 1,000 years or such other travel time as may be approved or specified by the Commission.

(b) On a case-by-case basis, the Commission may approve or specify some other radionuclide release rate, designed containment period or pre-waste-emplacement groundwater travel time, provided that the overall system performance objective, as it relates to anticipated processes and events, is satisfied. Among the factors that the Commission may take into account are:

(1) Any generally applicable environmental standard for radioactivity established by the Environmental Protection Agency;

(2) The age and nature of the waste, and the design of the underground facility, particularly as these factors bear upon the time during which the thermal pulse is dominated by the decay heat from the fission products;

(3) The geochemical characteristics of the host rock, surrounding strata and groundwater; and

(4) Particular sources of uncertainty in predicting the performance of the geologic repository.

60.130 General considerations.

Pursuant to the provisions of §60.21(c)(2)(i), an application to receive, possess, store, and dispose of high-level radioactive waste in the geologic repository operations area must include the principal design criteria for a proposed facility. The principal design criteria establish the necessary design, fabrication, construction, testing, maintenance, and performance requirements for structures, systems, and components important to safety and/or important to waste isolation. Sections 60.131 through 60.134 specify minimum requirements for the principal design criteria for the geologic repository operations area.

These design criteria are not intended to be exhaustive. However, omissions in §§60.131 through 60.134 do not relieve DOE from any obligation to provide such features in a specific facility needed to achieve the performance objectives.

60.131 General design criteria for the geologic repository operations area.

(a) *Radiological protection.* The geologic repository operations area shall be designed to maintain radiation doses, levels, and concentrations of radioactive material in air in restricted areas within the limits specified in part 20 of this chapter. Design shall include: ...

(2) Means to limit the time required to perform work in the vicinity of radioactive materials, including, as appropriate, designing equipment for ease of repair and replacement and providing adequate space for ease of operation;

(3) Suitable shielding;

(b) *Protection against design basis events.* The structures, systems, and components important to safety shall be designed so that they will perform their necessary safety functions, assuming occurrence of design basis events...

(c) *Protection against dynamic effects of equipment failure and similar events.* The structures, systems, and components important to safety shall be designed to withstand dynamic effects such as missile impacts, that could result from equipment failure, and similar events and conditions that could lead to loss of their safety functions.

(d) *Protection against fires and explosions.* (1) The structures, systems, and components important to safety shall be designed to perform their safety functions during and after credible fires or explosions in the geologic repository operations area.

(2) To the extent practicable, the geologic repository operations area shall be designed to incorporate the use of noncombustible and heat resistant materials.

(g) *Inspection, testing, and maintenance.* The structures, systems, and components important to safety shall be designed to permit periodic inspection, testing, and maintenance, as necessary, to ensure their continued functioning and readiness.

(h) *Criticality control.* All systems for processing, transporting, handling, storage, retrieval, emplacement, and isolation of radioactive waste shall be designed to ensure that nuclear criticality is not possible unless at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. Each system must be designed for criticality safety assuming occurrence of design basis events. The calculated effective multiplication factor (k_{eff})

must be sufficiently below unity to show at least a 5 percent margin, after allowance for the bias in the method of calculation and the uncertainty in the experiments used to validate the method of calculation.

(j) *Compliance with mining regulations.* To the extent that DOE is not subject to the Federal Mine Safety and Health Act of 1977, as to the construction and operation of the geologic repository operations area, the design of the geologic repository operations area shall nevertheless include provisions for worker protection necessary to provide reasonable assurance that all structures, systems, and components important to safety can perform their intended functions. Any deviation from relevant design requirements in 30 CFR, Chapter I, Subchapters D, E, and N will give rise to a rebuttable presumption that this requirement has not been met.

60.133 Additional design criteria for the underground facility.

(a) *General criteria for the underground facility.* (1) The orientation, geometry, layout, and depth of the underground facility, and the design of any engineered barriers that are part of the underground facility shall contribute to the containment and isolation of radionuclides.

(2) The underground facility shall be designed so that the effects of credible disruptive events during the period of operations, such as flooding, fires and explosions, will not spread through the facility.

(b) *Flexibility of design.* The underground facility shall be designed with sufficient flexibility to allow adjustments where necessary to accommodate specific site conditions identified through in situ monitoring, testing, or excavation.

(c) *Retrieval of waste.* The underground facility shall be designed to permit retrieval of waste in accordance with the performance objectives of § 60.111.

(d) *Control of water and gas.* The design of the underground facility shall provide for control of water or gas intrusion.

(e) *Underground openings.* (1) Openings in the underground facility shall be designed so that operations can be carried out safely and the retrievability option maintained.

(2) Openings in the underground facility shall be designed to reduce the potential for deleterious rock movement or fracturing of overlying or surrounding rock.

(h) *Engineered barriers.* Engineered barriers shall be designed to assist the geologic setting in meeting the performance objectives for the period following permanent closure.

(i) *Thermal loads.* The underground facility shall be designed so that the performance objectives will be met taking into account

the predicted thermal and thermomechanical response of the host rock, and surrounding strata, groundwater system.

60.135 Criteria for the waste package and its components.

(a) *High-level-waste package design in general.* (1) Packages for HLW shall be designed so that the in situ chemical, physical, and nuclear properties of the waste package and its interactions with the emplacement environment do not compromise the function of the waste packages or the performance of the underground facility or the geologic setting.

(2) The design shall include but not be limited to consideration of the following factors: 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.

IV. ACCEPTANCE CRITERIA AND GUIDANCE

General Guidance

In general, the License Application that will be submitted for construction authorization must provide sufficient information to enable NRC to make a determination that there is reasonable assurance that the types and amounts of radioactive waste described in the application can be received, possessed, and disposed of in the Yucca Mountain geologic repository operations area of the design proposed without unreasonable risk to the health and safety of the public. The engineered barriers, in conjunction with the natural barriers provided by the Yucca Mountain site, are expected to provide this reasonable assurance. Therefore, Chapter 6 of the License Application is the principal vehicle by which DOE will provide the information NRC reviewers need to understand the bases for the DOE conclusion that the engineered barrier system (excluding waste package) design is acceptable for licensing.

By definition in 10 CFR 60.2, the engineered barrier system means the waste packages and the underground facility. As noted in Section I, this chapter of the License Application discusses the engineered barrier system, excluding the waste packages. Therefore, this chapter addresses the underground facility portion of the engineered barrier system. According to 10 CFR 60.2, the underground facility means the underground structure, including openings and backfill materials, but excluding shafts, boreholes, and their seals. The underground structure is interpreted to include, in addition to openings and backfill, those systems, structures, and components whose functions include acting as barriers to transport of radionuclides.

To support the License Application and NRC reasonable assurance determination, this chapter must provide the following information:

- Description of the design bases that the engineered barrier system (excluding waste package) design is being designed to address

- Detailed description of the engineered barrier system and its components (excluding the waste package), including the design criteria that will enable the design bases to be successfully addressed

- Detailed analyses that support the conclusion that the engineered barrier system (excluding the waste package) design meets the design bases and addresses the applicable regulatory requirements, in particular the design criteria of 10 CFR 60.131 and 60.133 (YMP 1997a, Section 6.0)

Authors of Chapters 4, 5, and 6 should coordinate to ensure that consistency is maintained among the chapters, that unnecessary repetition is avoided, and that applicable engineered barrier system design criteria are addressed for all engineered barrier system components (YMP 1997a, Section 6.0). In addition, the Chapter 6 author should verify that the engineered barrier system design information presented in Chapter 6 is appropriately consistent with that in Chapters 7 and 8, which make use of the design information in performance assessments and analyses. Differences are expected, in particular because of assumptions on design parameters likely to be made in Chapters 7 and 8. These differences should be readily explainable.

In this chapter, it is not necessary to present an analysis of how the engineered barrier system supports meeting the subsystem performance objectives of 10 CFR 60.113 or the overall performance objective of 10 CFR 60.111. These performance objectives are addressed in Chapter 8 of the License Application. However, this chapter must describe how the design of the engineered barrier system (excluding the waste package) contributes to meeting these performance objectives. It must also demonstrate compliance with design criteria that support the system and subsystem performance objectives, that is, those applicable design criteria in 10 CFR 60.131 and 60.133. It must also show compliance with DOE-developed design criteria that have been developed to implement the regulatory design criteria.

The engineered barrier system (excluding the waste package) may provide some degree of shielding of personnel from neutron and gamma radiation emissions from the emplaced waste, depending on the configuration of the engineered barrier system and the layout of the underground facility. If applicable, discuss this issue in the appropriate subsections of Section 6.2 with regard to compliance with radiological protection requirements of 10 CFR 60.131(a)(3). Such shielding provided by components of the engineered barrier system may or may not allow approach by personnel without the presence of additional shielding for personnel. Additional shielding of personnel will likely be needed for loading and waste emplacement operations, including placement of certain components. The need for additional shielding is discussed in Chapters 4, 9, and 10 of the License Application. Discussions of shielding in Chapter 6 serve largely as an input to shielding evaluations in those chapters.

Chapter 6 of the License Application should be written in substantial detail because the engineered barrier system is "first-of-a-kind." This level of detail should equal or exceed that provided for safety-related structures and components in a nuclear power plant Final Safety Analysis Report. At the time of the submittal of the License Application, the engineered barrier system (excluding the waste package) will still be a design and not a constructed configuration item. It is therefore the burden of DOE as applicant to provide analyses of the performance of

the engineered barrier system (excluding the waste package) design with uncertainties sufficiently low to enable NRC to make its reasonable assurance findings.

It should also be stated that it is counterproductive to provide more detail in the License Application beyond that needed to support the NRC reasonable assurance findings. Excessive details result in excess information in the docket and the need for excessive number of changes in the Safety Analysis Report after its original submittal. Therefore, the author must provide sufficient information to support the licensing/safety case and to show compliance with regulations, and generally nothing more.

It is important in the License Application to identify NRC regulatory guides and industry standards with which design elements are consistent, because consistency with these industry-accepted and often NRC-accepted documents eases the burden of the regulator in reviewing the adequacy of the design. This is the case because the regulator, rather than needing to determine whether the method and its application are correct, need only determine that use of the standard or guide is appropriate and that the applicant has correctly applied it. Unfortunately, because of the unique time frame of regulatory concern for geologic disposal, standards for the postclosure period do not exist. Where possible, regulatory guides and industry standards will be applied, as determined by the project's compliance program and as discussed in this technical guidance document chapter. However, many aspects of engineered barrier system (excluding the waste package) performance will need to be addressed without reference to standards. These aspects will need to be addressed in much more detail in the License Application and its references than will other aspects of design for which relevant standards exist.

Most of the guidance in this chapter is based on, or extracted from, these documents:

- NUREG-1323, *License Application Review Plan for a Geologic Repository for Spent Nuclear Fuel and High-Level Radioactive Waste* (NRC 1995)
- YMP/97-01Q, *Management Plan for the Development of a Project Integrated Safety Assessment (PISA)* (YMP 1997a)
- NUREG-1567, *Standard Review Plan for Spent Fuel Dry Storage Facilities* (NRC 1996a)
- Regulatory Guide 3.61, *Standard Format and Content for a Topical Safety Report for a Spent Fuel Dry Storage Cask* (NRC 1989a)
- NUREG-1536, *Standard Review Plan for Dry Cask Storage Systems* (NRC 1997a)

NUREG-1323 (NRC 1995) is largely incomplete and will become inapplicable to Yucca Mountain upon the expected issuance of a new 10 CFR 63 as discussed in Section III. Information from NUREG-1323 considered likely to be applicable under the new regulation is included in the technical guidance document, though the NUREG is not referenced as an authoritative source for such information. YMP/97-01Q (YMP 1997a) contains the Project Integrated Safety Assessment Content Guide that has been developed by the Yucca Mountain Project based primarily on the NRC Regulatory Guide DG-3003, *Format and Content for the*

License Application for the High-Level Waste Repository (NRC 1990). The Project Integrated Safety Assessment Content Guide is the starting point for the development of this technical guidance document for License Application Preparation. As is the case for NUREG-1323, DG-3003 will become inapplicable to Yucca Mountain upon the expected issuance of a new 10 CFR 63 as discussed in Section III. Information from DG-3003 considered likely to be applicable under the new regulation is included in the technical guidance document, though DG-3003 is not referenced as an authoritative source for such information. NUREG-1567 (NRC 1996a), NUREG-1536 (NRC 1997a), and Regulatory Guide 3.61 (NRC 1989a), though not directly applicable to the repository or engineered barrier system, pertain to containers for dry storage of spent nuclear fuel and therefore contain much useful and transferable information for developing both Chapter 5 and Chapter 6, of the License Application for the repository. These documents contain the type of detailed information that is largely missing from NRC guidance documents for the repository. The dry storage documents also reference other NRC and industry guidance, and these documents are cited in this technical guidance document, as appropriate.

NRC has issued numerous regulatory guides regarding fabrication and installation of safety-system components. Some are discussed in this general guidance section of the chapter, as they are potentially useful for engineered barrier system (excluding the waste package) design in general. Others are more specific and are discussed in the subsections of this chapter to which they most directly apply. Some of the references in this chapter to NRC regulatory guidance documents (as well as NUREGs, branch technical positions, and standard review plans) have been made based on preliminary evaluation only, in which case the guidance will be worded using phrases such as "potentially applicable." As the compliance program continues to analyze regulatory guides, this guidance will be revised as appropriate. It should be further noted that many of the following regulatory guides apply to structures fabricated from materials such as metals and concrete. It is not known at this time to what extent the engineered barrier system, excluding the waste package, will contain these materials, or whether these guides would apply to their application in the engineered barrier system (excluding the waste package). Final determination on the issue of applicability of these guidance documents to the project will be made in the future via the compliance program.

The following regulatory guides may ultimately be determined to be applicable to the design of the engineered barrier system (excluding the waste package). Note that it may not be necessary to discuss each regulatory guide listed below in each subsection. Instead, where a document applies throughout the systems or subsystems described in the chapter (assuming some of the regulatory guides stated below are applicable and are to be discussed in the License Application), make a global statement in the appropriate location (such as under design criteria) regarding how the project has addressed each applicable regulatory guide with regard to engineered barrier system (excluding the waste package) design.

- Regulatory Guide 1.31, *Control of Ferrite Content in Stainless Steel Weld Metal* (NRC 1977a), describes a method acceptable to NRC for implementing requirements with regard to the control of welding in fabricating and joining safety-related austenitic stainless steel components and systems.

- Regulatory Guide 1.34, *Control of Electroslag Weld Properties* (NRC 1972), describes an acceptable method of implementing these requirements with regard to the control of weld properties when fabricating electroslag welds for nuclear components made of ferritic or austenitic materials.

- Regulatory Guide 1.43, *Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components* (NRC 1973c), describes controls for underclad weld cracking when cladding low-alloy steel with stainless steel.

- Regulatory Guide 1.50, *Control of Preheat Temperature for Welding of Low-Alloy Steel* (NRC 1973b), describes an acceptable method of implementing General Design Criterion 1, "Quality Standards and Records," of Appendices A and B to 10 CFR 50 requirements with regard to the control of welding for low-alloy steel components during initial fabrication.

- Regulatory Guide 1.71, *Welder Qualification for Areas of Limited Accessibility* (NRC 1973e), specifies welder qualification for restricted spaces.

- Regulatory Guide 1.84, *Design and Fabrication Code Case Acceptability ASME Section III Division 1* (NRC 1994a), describes applicable ASME design and fabrication code cases with NRC annotations.

- Regulatory Guide 1.85, *Materials Code Case Acceptability ASME Section III Division 1* (NRC 1994b), provides applicable ASME materials code cases with NRC annotations.

- Regulatory Guide 1.94, *Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants* (NRC 1976b), describes installation, inspection, and testing of structural concrete and structural steel during the construction phase.

- Regulatory Guide 1.116, *Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment* (NRC 1976a), provides guidance for installation, inspection, and testing of equipment.

- Regulatory Guide 1.120, *Fire Protection Guidelines for Nuclear Power Plants* (NRC 1977b), implements the philosophy of defense-in-depth protection against the hazards of fire and its associated effects on safety-related equipment.

- Regulatory Guide 1.130, *Service Limits and Loading Combinations for Class 1 Plate-And-Shell-Type Component Supports* (NRC 1978b), specifies the service limits and combinations of loadings associated with normal operation, postulated accidents, and specified seismic events for the design of Class I plate-and-shell-type component supports as defined in Subsection NF of Section III of ASME *Boiler and Pressure Vessel Code* (ASME 1989).

- Regulatory Guide 1.136, *Materials, Construction, and Testing of Concrete Containments* (NRC 1981a), provides guidance for materials control and control of special processes, such as welding, and proper testing to be performed with regard to the materials, construction, and testing of concrete containments.

- Regulatory Guide 1.150, *Ultrasonic Testing of Reactor Vessel Welds During Pre-Service and In-Service Inspections* (NRC 1983), describes procedures acceptable to the NRC staff for implementing 10 CFR 50 criteria with regard to preservice and inservice examinations of reactor vessel welds.

- Regulatory Guide 3.29, *Preheat and Interpass Temperature Control for the Welding of Low-Alloy Steel for Use in Fuel Reprocessing Plants and in Plutonium Processing and Fuel Fabrication Plants*, (NRC 1975a), provides information on welding of low-alloy steel.

- Regulatory Guide 3.9, *Concrete Radiation Shields* (NRC 1973a), discusses the requirements and recommended practices contained in ANSI N101.6-1972, *Concrete Radiation Shields* (ANSI/ANS 1972), that are acceptable for the construction of radiation shielding structures for hot laboratories, radiochemical plants, experimental facilities, and nuclear fuel fabrication plants subject to certain exceptions discussed in the guide.

YMP/94-05, *Mined Geologic Disposal System License Application Annotated Outline* (DOE Office of Civilian Radioactive Waste Management, in draft), contains potentially useful sample text for the License Application. Review Chapter 5, and extract appropriate and useful text, recognizing that the document is unapproved and that some of the concepts and information may be outdated. Therefore, any text lifted from that document will need to be verified to be up-to-date in light of the current engineered barrier system (excluding the waste package) design concepts.

6.0 PURPOSE, SUMMARY, AND CONCLUSIONS

Provide the purpose of Chapter 6 of the License Application for construction authorization, which is to provide a detailed description of the engineered barrier system (excluding the waste package) and evaluate compliance with design criteria to support the radiological safety case and waste isolation. NRC views these regulatory design criteria as an approach to providing adequate safety margin and defense in depth against undesirable and possible unpredicted preclosure and postclosure conditions, events, and consequences. State that each section in this chapter addresses how the information contained therein relates to the demonstration of compliance with the design criteria in 10 CFR 60.131 and 60.133 (YMP 1997a, Section 6.0).

State that by regulatory definition in 10 CFR 60.2, the waste package and the waste form are a part of the engineered barrier system. Clearly delineate the boundary between the waste package, which is discussed in Chapter 5, and the remainder of the engineered barrier system, which is discussed in this chapter. Refer the reader to Chapter 5 of the License Application for discussion of the waste forms and design of the waste package. Also, according to the regulatory definition in 10 CFR 60.2, the engineered barrier system includes the underground facility. List

those structures, systems, and components of the underground facility considered to be part of the engineered barrier system. These include, as required by the regulations, underground openings and backfill materials. They also include those structures, systems, and components of the underground facility whose functions include contributing to the isolation of radioactive materials. The compliance program (CRWMS M&O 1998) discusses parts of the engineered barrier outside the waste package as the "ex-container" system; this system does not include the ground support system. For convenience, therefore, the ground support system will be discussed in Chapter 4 of the License Application.

Changes in the near-field environment in response to thermal loads are discussed in Sections 3.7 and 4.4 of the License Application; references to these sections should be provided (YMP 1997a, Section 6.0).

Discuss the organization of Chapter 6. Summarize the role of the information presented in Chapter 6 as it relates to the ability of a repository to operate safely. Explain how the engineered barrier system (excluding the waste package) design supports the DOE safety case. State that detailed discussion of the approach to waste containment and isolation is provided in Chapter 8 of the License Application. Describe in general terms the relationship between the engineered barrier system (excluding the waste package) design, described in Chapter 6, and the repository preclosure radiological safety assessment and postclosure performance assessment, which are described in Chapters 7 and 8, respectively (YMP 1997a, Section 6.0).

Briefly describe, at a summary level, the design features of the engineered barrier system (excluding the waste package) and its components, and how they are effective against the release of radioactive material to the environment in accordance with 10 CFR 60.21(c)(1)(ii)(D), 10 CFR 60.31(a)(1)(v), 10 CFR 60.31(a)(2), and 10 CFR 60.133(h). Relate the design features to the design bases. Include a general description of the characteristics and materials. Provide a brief description of how each component contributes to the overall engineered barrier system (excluding the waste package) design and to the acceptable performance of the isolation system. Summarize the evaluation results in this summary section, with reference made to the more detailed discussions in the appropriate subsections. If the functional adequacy of any aspect of the engineered barrier system (excluding the waste package) design has not been fully demonstrated through engineering analysis and/or physical testing of prototypes, identify those aspects in this section and provide an explanation and schedule for resolution of the issue prior to receipt of radioactive materials at the repository. Provide a cross-reference to Section 11.11 of the License Application, where a list of all such incomplete information for the repository is provided (NRC 1996a, Section 7.4.2.1). Current project plans are to minimize the amount of such incomplete information at the time of License Application submittal.

Identify the subissues of the NRC key technical issues (KTIs) that are addressed in this chapter. Refer to the more detailed discussions in identified sections/subsections that follow for information regarding status of the subissues and for discussion of the acceptance criteria applicable to this chapter. M&O Licensing can provide copies of the reports for KTI information regarding how DOE has dispositioned and/or responded to each related issue resolution status report (IRSR). Relate this discussion to compliance with 10 CFR 60.133(i).

6.1 ENGINEERED BARRIER SYSTEM ENVIRONMENT SUMMARY DESCRIPTION

Provide a general description of the expected engineered barrier system environment, describing those portions of the geologic repository operations area within which the engineered barrier system (excluding the waste package) will be installed, the interface between the engineered barrier system (excluding the waste package) and the remainder of the geologic repository operations area, and the conditions at the interface. Describe the relationship of the engineered barrier system environment to overall repository performance. Show how the various aspects of the engineered barrier system environment, including both the natural conditions of the geologic setting and the man-made conditions and underground structures, support the design criteria in 10 CFR 60.131 and 60.133, contain engineered features specifically designed to meet the design criteria, or contain limitations that require engineered remediation by the engineered barrier system (excluding the waste package) (YMP 1997a, Section 6.0).

Provide an overview of the major site features that affect geologic repository operations area design and performance as required by 10 CFR 60.21(c)(1). Limit the discussion to those site features that affect the engineered barrier system (excluding the waste package). Note that 10 CFR 60.21(c)(1) also requires "appropriate attention" to features of the site that might affect geologic repository operations area design and performance. Describe both the pre-emplacement and post-emplacement environments. Focus on the interactions of the environment with the engineered barrier system (excluding the waste package) and on the effects of the environment on the ability of the engineered barrier system (excluding the waste package) 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; and water chemistry and water flow rate. Provide a reference to Chapter 3 for additional information on these environmental topics (YMP 1997a, Section 6.1).

6.2 ENGINEERED BARRIER SYSTEM (EXCLUDING WASTE PACKAGE) DESIGN

This section describes the principal design features and performance of the engineered barrier system (excluding the waste package) and all components, except for the waste package, which is described in Chapter 5. In this introductory subsection, briefly discuss the overall design features of the engineered barrier system (excluding the waste package) to address 10 CFR 60.21(c)(1)(ii)(D), 10 CFR 60.31(a)(1)(v), 10 CFR 60.31(a)(2), and 10 CFR 60.133(h). Summarize and discuss the adequacy of the overall engineered barrier system (excluding the waste package) performance evaluation (NRC 1989a).

6.2.1 Design Bases

Design bases are defined in 10 CFR 60.2 as follows:

> That information that identifies the specific functions to be performed by a structure, system, or component 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 accident for which a structure, system, or component must meet its functional goals.

Discuss the design bases for that portion of the underground facility, including emplacement drifts and backfill in them (if used), emplacement drift inverts, and other items that are defined in the project's system description documents to be part of the engineered barrier system (excluding the waste package). Organize the design bases by component. Include the overall purpose and function of the engineered barrier system (excluding the waste package) as part of the design basis discussion.

Describe the purpose of the information in this section as it relates to the overall repository performance and the design criteria in 10 CFR 60.131 and 60.133. Show how the underground facility design acts in tandem with the waste package design to form engineered barriers that assist the geologic setting in meeting the performance objectives and environmental constraints for the postclosure period to show compliance with 10 CFR 60.133(h) . Also, relate these design bases to system constraints and design basis events (YMP 1997a, Section 6.2.1).

For each component, describe the design bases, if feasible, in the same order as the design itself is presented in the sections of the License Application that follow. The order should then be:

- General design bases and design bases that do not fall into the categories that follow
- Design bases related to materials
- Design bases related to thermal performance
- Design bases related to criticality
- Design bases related to structural performance
- Design bases related to shielding

Additional regulatory guidance for some of these design bases discussions is provided in the paragraphs below.

To the extent that the engineered barrier system design bases are related to criticality control, describe safety margins to ensure that, in the preclosure period, a subcritical condition exists under all credible conditions. For the postclosure period, discuss design bases that provide defense in depth against the probability and consequences of criticality for the period of regulatory concern. Describe the criticality source term assumptions on which the design of the waste package and engineered barrier system are based. Cross-reference Chapter 5 as necessary, as most of the criticality discussion is expected to appear in that chapter.

For the design bases related to the engineered barrier system (excluding the waste package) structural design, include load combinations for accidents and design basis events. Provide a table showing load combinations related to allowed stresses. The engineered barrier system (excluding the waste package) must maintain its functional integrity under preclosure accident conditions. If the engineered barrier system (excluding the waste package) will be degraded by design basis events (which only apply to the preclosure period), state here and demonstrate in Section 6.2.3, that DOE will develop requirements and procedures for determination and correction of the degradation, or other acceptable remedial action. The description of such plans

in the License Application will need to be sufficient to ensure NRC that DOE will have the means to correct any degradation caused by a design basis event (NRC 1996a, Section 7.4.2.1).

Discuss the seismic design bases used in the design and installation of the engineered barrier system (excluding the waste package). Provide the seismic classification of systems, structures, and components of the engineered barrier system (excluding the waste package). This information may be presented in the form of text, tables, or drawings, but must show classification of specific items. Explain the basis for the selection of these items and demonstrate that the classification of structures, systems, and components supports design bases for protection against seismic events (NRC 1978a). Include input criteria for seismic design. Justify the seismic response spectra used and present the earthquake time functions or other data from which they were derived. Reference *Topical Report: Preclosure Seismic Design Methodology for a Geologic Repository at Yucca Mountain* (YMP 1997b) and the future topical report on preclosure seismic design bases for the source of the seismic design bases. For damping values used in the design, provide a comparison of the response spectra derived from the time history and the design response spectra. Identify the system period intervals at which the spectra values were calculated.

Discuss design bases related to the engineered barrier system (excluding the waste package) shielding functions, if provided. Shielding provided by the engineered barrier system (excluding the waste package) is strictly a preclosure issue, because shielding will not be required for the postclosure period in which personnel access to the repository will not be possible. Discuss whether the shielding function may be acceptably degraded by a design basis event. Radiation shielding provided by the engineered barrier system (excluding the waste package) and credited in analyses of radiation protection for the repository operating personnel must not degrade during normal or off-normal preclosure conditions. State that the loss of any shielding function must be readily apparent after a design basis event. Proposed procedures for testing shielding effectiveness after a design basis event must be discussed in the License Application in sufficient detail to demonstrate feasibility (NRC 1996a, Section 7.2.4.1).

6.2.2 Design Description

Provide in this section the general and detailed design criteria that govern the design as required by 10 CFR 60.21(c)(2),10 CFR 60.31(a)(1)(iii), and 10 CFR 60.130, and describe the design itself.

Design Overview

First, provide a brief overview description of the design of the engineered barrier system that is being developed from these design criteria, except for the waste package which is described in Chapter 5. Discuss major engineered barrier system (excluding the waste package) components and their intended functions and relations in the overall repository design.

General Design Criteria

Describe the general design criteria that govern the engineered barrier system (excluding the waste package) design, using the technical criteria of Subpart E to 10 CFR 60 and using a similar

design philosophy as that contained in the reactor General Design Criteria of 10 CFR 50, Appendix A. Such criteria should relate to topics such as the following:

Control of Releases of Radioactive Materials to the Environment–Briefly summarize how the engineered barrier system (excluding the waste package) design contributes to controlling the release of radionuclides to the environment.

Design for Protection Against Natural Phenomena–Provide a statement that the engineered barrier system (excluding the waste package) is designed to withstand credible natural phenomena, including a brief explanation of how the design provides this protection.

Redundancy and Independence–If the engineered barrier system (excluding the waste package) design provides an independent and redundant function regarding, for example, containment of radionuclides, state how the design provides for independence and redundancy. To the extent that the engineered barrier system (excluding the waste package) provides system independence or redundancy, state that the design can withstand the effects of natural phenomena or accidents without reducing its redundant function and explain how it does so.

Residual Heat Removal–Describe how the engineered barrier system (excluding the waste package) design supports or provides for residual heat removal from the waste package such that maximum internal temperatures of the waste package are not exceeded.

Control of Criticality–To the extent that it can be shown that the engineered barrier system (excluding the waste package) limits the amount of water in contact with the waste package and thus provides a criticality-related safety function, describe how the engineered barrier system (excluding the waste package) helps prevent criticality. Also describe how the engineered barrier system (excluding the waste package) contributes to control of postclosure criticality.

Specific Design Criteria

Provide the principal specific design criteria applicable to the engineered barrier system (excluding the waste package), using the project's system description documents as the reference. Explain the relationship between the design bases and design criteria, discussed in Section 6.2.1, as well as the relationship between the design criteria and the regulatory performance objectives and criteria in 10 CFR 60. Design criteria related to the specific aspects of the design that are discussed in the sections that follow may be presented in those sections rather than here. As discussed further below, divide the design criteria discussion for each component into preclosure and postclosure criteria.

Codes and Standards–To comply with 10 CFR 60.21(c)(2)(iv), list in a table and explain how industry codes and standards have been or will be used in the design, including installation procedures and practices. Include any standards regarding compaction of backfill, selection, testing, and installation of materials, such as grouts and concretes. Recognize that NRC has not generally accepted use of a set of criteria selected from multiple standards unless it can be shown that the selected criteria meet the most limiting requirements of each code. Should such a mixed set of criteria be proposed, provide this justification (NRC 1997a, Section 3.V.2.b.i(2)). (NRC

has accepted use of different codes for different components, when one is important to safety and the other not.)

Preclosure Design Criteria–Discuss aspects of the engineered barrier system (excluding the waste package) design that will help ensure there will be no uncontrolled release of radioactivity from the waste package during the preclosure period. Describe criteria for protection against any radioactive releases from postulated off-normal operations, internal change, or external natural phenomena. Describe criteria selected for backup containment [that is, containment of radionuclides should the primary (waste package) barrier fail] and delineate the extent to which the engineered barrier system (excluding the waste package) design contributes to achieving the lowest practical level of radioactive releases from the waste package (NRC 1989a).

Address the following design criteria:

- The engineered barrier structures, systems, and components important to safety shall be designed so that natural phenomena and environmental conditions anticipated at the geologic repository operations area will not interfere with necessary safety functions.

- The engineered barrier system (excluding the waste package) structures, systems, and components shall be designed so that operations in the underground facility can be carried out safely and the waste will be retrievable for the designated period of time.

In the design criteria discussion, include a description of the design criteria that address the natural phenomena and events to which the engineered barrier system (excluding the waste package) might be subjected. Ensure that design criteria described in this chapter that were derived from accident analyses or other analyses are consistent with those analyses (NRC 1996a, Section 3.5.3). State that the engineered barrier system (excluding the waste package) is protected against flooding through repository access ramps and shafts because all the ramps and shafts leading to the underground facility will be located above the probable maximum flood level.

Describe how the engineered barrier system (excluding the waste package) design supports drainage of water from the drifts. Describe the vulnerability of engineered barrier system (excluding the waste package) structures, systems, and components to flood damage from buildup in the emplacement drifts of water from percolation through the rock or from human activities, and describe the design criteria that protect these items against damage (NRC 1989a). Provide design criteria selected to ensure that all safety functions will successfully withstand credible fire and explosion conditions as required by 10 CFR 60.131(d)(1). Also, provide design criteria that incorporate, to the maximum extent feasible, use of noncombustible and heat-resistant materials to address 10 CFR 60.131(d)(2).

Provide the design criteria that will support the ability of the engineered barrier system (excluding the waste package) structures, systems, and components important to safety to withstand dynamic effects such as missile impacts that could result from equipment failure or personnel error, and similar events and conditions that could lead to loss of their safety functions. Include consideration of the potential impact on safety-related components of failure or

degradation of nonsafety-related components caused by missile impacts. Show that nonsafety-related components whose failure or degradation from missile impacts could affect safety-related components are properly protected against such impacts. These criteria address 10 CFR 60.131(c). State the manner in which protection is provided (for example, physical separation of redundant components, shields or barriers, etc.). Describe the method for identifying potential missile sources and defend the adequacy of that method.

Also, provide the design criteria that will ensure flexibility of the engineered barrier system (excluding the waste package) design to allow adjustments where necessary to accommodate specific site conditions identified through in situ monitoring, testing, or excavation as required by 10 CFR 60.133(b).

Provide the design criteria for the engineered barrier system (excluding the waste package) that contribute to the control of water and gas intrusion as required by 10 CFR 60.133(d).

Discuss consideration given in the engineered barrier system (excluding the waste package) design criteria to closure of the repository, retrieval of the waste for inspection, or retrieval from the repository in accordance with 10 CFR 60.133(c) and 10 CFR 60.133(e)(1) (NRC 1989a).

Describe design criteria for engineered barrier system (excluding the waste package) structures, systems, and components in underground openings that reduce the potential for deleterious rock movement or rock fracturing, as required by 10 CFR 60.133(e)(2).

Postclosure Design Criteria–Using the criteria provided by the system description documents, list the design criteria applicable to the postclosure period. A detailed description of criteria that impact the engineered barrier system's function of retarding release of radionuclides from the repository should be provided. Provide the design criteria that assist the geologic setting in meeting the performance objectives for the period following permanent closure in accordance with 10 CFR 60.133(h). These are the criteria that support the waste package in meeting its design criteria in 10 CFR 60.135(a) regarding adverse interactions with the environment, the overall performance objectives of 10 CFR 60.112 to ensure that releases of radioactive materials conform to U.S. Environmental Protection Agency standards, and the containment and release rate requirements of 10 CFR 60.113(a).

Detailed Design Description

Provide the following information for the engineered barrier system (excluding the waste package) components as applicable. The engineered barrier system (excluding the waste package) will include fabricated components and may include fill-type materials. The type of design information provided will depend on whether a given component is fabricated or fill.

Provide a detailed, non-proprietary description of the design of the engineered barrier system (excluding the waste package) and each of its components. Summarize the principal characteristics of the engineered barrier system (excluding the waste package). Include (as needed to support the licensing case) the materials of construction, special materials such as radionuclide adsorbers, dimensions, weight, internal structures, means of passive heat dissipation, amount of shielding provided, and mechanisms contributing to containment.

Describe properties of the engineered barrier system (excluding the waste package) that affect its interactions with the waste package components. All these descriptions must be described in sufficient detail to enable NRC reviewers to evaluate their effectiveness (NRC 1997a, Section 1.III.2). Provide no additional detail beyond this level. Specifically, avoid adding construction level detail. For example, if certain weights or dimensions are not needed for NRC to determine the acceptability of the design, do not include them in the License Application.

Identify structures, systems, and components of the engineered barrier system (excluding the waste package), and indicate whether or not they have been classified as important to radiological safety as required by 10 CFR 60.21(c)(1)(ii)(E) or as important to waste isolation. Identify structures, systems, and components that are expected to be required to maintain the capability to retrieve the waste if necessary as required by 10 CFR 60.21(c)(12), 60.111(b), and 60.133(c). Refer the reader to Subsection 4.3.5 of the License Application for a discussion of plans for waste retrieval as required by 10 CFR 60.21(c)(12). During the preclosure period, and to the extent that the engineered barrier system (excluding the waste package) may be susceptible to design basis events, provide performance requirements, including margins of safety under design basis events that satisfy 10 CFR 60.21(c)(3) and 10 CFR 60.131(b) requirements.

Provide the quality assurance classification of all structures, systems, and components in the engineered barrier system (excluding the waste package). State that all such items have been or will be designed and constructed in accordance with an NRC-approved quality assurance program, namely that provided in DOE/RW-0333P (DOE 1998). Provide a cross-reference to Chapter 2, where the method for classification is described.

State that instrumentation installed in the engineered barrier system (excluding the waste package) for use during the preclosure period (if any such equipment is installed) will have a safety classification commensurate with its function, and justify the classification. Describe such equipment in sufficient detail to support in-depth NRC staff evaluation (NRC 1997a, Section 4.V.2.b). Refer the reader to Chapter 12 of the License Application for discussion of instrumentation related to performance confirmation.

Identify the effects on engineered barrier system (excluding the waste package) performance of structures, systems, and components that are neither important to safety nor important to waste isolation (NRC 1997a, Section 2.III.1).

Provide design drawings, including overall and cutaway sketches, and specifications that clearly show the safety features considered in the safety analysis and that are sufficiently detailed to allow NRC reviewers to evaluate the effectiveness of the engineered barrier system (excluding the waste package) and its components. Drawings should be complete enough that NRC can ascertain the major design of the engineered barrier system (excluding the waste package) and its general operations and determine its acceptability for licensing. If proprietary concerns will exist beyond the issuance of the construction authorization, sketches, drawings, and diagrams may be made non-proprietary by depicting less detail or by illustrating generic components that fulfill the design function but differ from the actual design. However, these representations should show the operational concept and safety-related features in sufficient detail to form an acceptable basis for public review and comment. Ensure that dimensions, materials, and other drawing

details are consistent with the same information presented in the chapter text, and that all information in the drawings is legible (NRC 1997a, Sections. 1.III.2, 1.IV.2, 1.V.1, and 1.V.2).

Identify the value of design parameters used to meet the design criteria previously described in this section. Describe uncertainties associated with the parameters and the treatment of those uncertainties.

Describe the process for construction of the engineered barrier system (excluding the waste package) and describe operations that will occur in its vicinity. Discuss operating limitations, precautions, or other steps that will prevent or minimize the likelihood of damage to the engineered barrier system (excluding the waste package) during such operations (NRC 1997a, Section 2.V.2).

At a summary level, describe alternative design features considered, and the comparative evaluations related to the selected design features. In particular, provide a discussion to support the selected alternatives for equipment important to waste containment and isolation to show compliance with 10 CFR 60.21(c)(1)(ii)(D). Cross-reference where these alternative design analyses are discussed in detail (Section 6.2.3 that follows).

In accordance with 10 CFR 60.131(g), the repository design must permit periodic inspection, testing, and maintenance of structures, systems, and components important to safety. Describe periodic testing and maintenance expected to be needed during the preclosure period to ensure the continued satisfactory performance of the engineered barrier system (excluding the waste package). Describe the aspects of the engineered barrier system (excluding the waste package) design that permit the necessary maintenance and testing, paying particular attention in the discussion to any parts of the engineered barrier system (excluding the waste package) that are important to safety.

To the extent engineered barrier system (excluding the waste package) structures, systems, and components could affect personnel safety, demonstrate that the personnel safety requirements of 30 CFR Chapter I will be met by the engineered barrier system (excluding the waste package) design, or show that equivalent protection is required in accordance with 10 CFR 60.131(j).

Testing–Discuss engineered barrier system (excluding the waste package) acceptance criteria and the bases for the acceptance criteria. Testing for specific aspects of the design discussed in the subsections that follow may be discussed in those subsections instead of here. Discuss analyses or tests to be performed to show the acceptance criteria are met. Discuss visual or other 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 (NRC 1989a).

Discuss analyses and/or tests for engineered barrier system (excluding the waste package) components that are important to safety. If a characteristic such as longevity cannot be tested, justify an upper limit. Discuss acceptance criteria and actions to be taken, such as replacement, if the criteria are not met (NRC 1989a).

State that any component whose failure would impair engineered barrier system (excluding the waste package) effectiveness will be, or has been, analyzed or tested under the most severe conditions for which it was designed. State and support the position that tests performed are expected to ensure adequate representation of those conditions that would prevail if the actual system were in use. State that the manufacturers and suppliers of components important to safety will maintain a quality assurance program adequate to ensure that acceptance testing of a given engineered barrier system (excluding the waste package) component is equivalent to acceptance testing of all components supplied by that manufacturer (NRC 1989a).

Computer Code Validation and Verification–Identify the verification and validation status of all computer codes used in the design and analysis of structures, systems, and components discussed in this chapter and subject to requirements of DOE/RW-0333P, *Quality Assurance Requirements and Description* (DOE 1998).

Qualification of Data–Identify the qualification status of design input data required to be qualified.

Peer Review–Identify the extent of compliance of peer review results used as design input with NUREG-1297, *Peer Review for High-Level Nuclear Waste Repositories* (NRC 1988).

Expert Elicitation–Identify the extent of compliance of expert elicitation results used as design input with NUREG-1563, *Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program* (NRC 1996b).

Engineered Barrier System (Excluding the Waste Package) Materials, Thermal, Structural, Criticality, and Shielding–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 (Excluding the Waste Package) Materials Design

Provide the materials design criteria for the engineered barrier system (excluding the waste package), as described in the project system description documents.

Describe the engineered barrier system (excluding the waste package) materials used. Discuss the material alternatives considered. Identify the materials specifications and general manufacturing and emplacement methods to be used. Discuss any applicable industry codes and standards that were used in the material design. Identify the value of design parameters used to meet the design criteria. Describe any uncertainties associated with the parameters and the treatment of those uncertainties.

Provide plan and cross-section drawings of the engineered barrier system (excluding the waste package), showing dimensions and tolerances and highlighting materials-related features as necessary to support the licensing case. Explain why the materials-related design features were chosen. Describe potential alternative measures, and explain why they were not chosen, as required by 10 CFR 60.21(c)(1)(ii)(D).

Identify and state the properties of any material to be used for embedments, inserts, conduits, pipes, or other items to be embedded in concrete engineered barrier system (excluding the waste package) components. NUREG-1536 (NRC 1997a) states that all such embedments must satisfy the requirements of the code used in designing the reinforced concrete structure in which they are embedded. It further states that aluminum should not be used for any embedded objects that will be in contact with wet concrete because of the potential for concrete degradation from an adverse chemical reaction. State, therefore, that all embedments meet the code for the material in which they are embedded and that no aluminum embedments will be allowed to come in contact with wet concrete. If aluminum is used, explain the controls that will prevent its contact with wet concrete (NRC 1997a, Section 3.V.c.ii).

State how or whether the engineered barrier system (excluding the waste package) materials will affect waste package corrosion and/or material sensitization to corrosion and provide evidence supporting the statements. If the engineered barrier system (excluding the waste package) reduces the contact of fluoride, chloride, and dissolved oxygen ions with the waste packages, so state. Regulatory Guide 1.44, *Control of Use of Sensitized Stainless Steel* (NRC 1973d), provides guidance and information on protection against contaminants capable of causing stress corrosion cracking during fabrication, shipment, storage, construction, testing, and operation of stainless steel components and systems. Regulatory Guide 3.37, *Guidance for Avoiding Intergranular Corrosion and Stress Corrosion in Austenitic Stainless Steel Components of Fuel Reprocessing Plants* (NRC 1975b), provides additional guidance and information on the subject of stainless steel corrosion and sensitization. The project will assess the applicability of these guidance documents, and appropriate instruction for the License Application author will be provided in a future revision to this technical guidance document.

6.2.2.2 Engineered Barrier System (Excluding the Waste Package) Thermal Design

Provide the thermal design criteria for the engineered barrier system (excluding the waste package), as described in the project system description documents.

Describe aspects of the engineered barrier system (excluding the waste package) design and its components related to thermal performance. Discuss any applicable industry codes and standards that were used in the thermal design. (Cross-reference the list provided earlier in Section 6.2.2 to avoid repetition.) Identify the value of design parameters used to meet the design criteria. Describe uncertainties associated with the parameters and the treatment of those uncertainties. Include specifications for engineered barrier system (excluding the waste package) components related to thermal performance (NRC 1989a).

Provide detailed descriptions of the thermal-related engineered barrier system (excluding the waste package) design features. Include dimensions, material descriptions, and manufacturing processes as necessary to support the licensing case. Provide plan and cross-section drawings of the engineered barrier system (excluding the waste package), showing dimensions and tolerances and highlighting thermal-related features as necessary to support the licensing case. Explain why the thermal-related design features were chosen. Describe potential alternative measures, and explain why they were not chosen, as required by 10 CFR 60.21(c)(1)(ii)(D).

6.2.2.3 Engineered Barrier System (Excluding the Waste Package) Criticality Design

Provide the criticality design criteria for the engineered barrier system (excluding the waste package), as described in the project system description documents.

Describe aspects of the design of the engineered barrier system (excluding the waste package) and its components related to criticality control. Identify the value of design parameters used to meet the design criteria. Describe uncertainties associated with the parameters and the treatment of those uncertainties. Discuss any applicable industry codes and standards that were used in the criticality design. (Cross-reference the list provided earlier in Section 6.2.2 to avoid repetition.) Explain any criticality-related design functions.

Describe how the engineered barrier system (excluding the waste package) design is expected to support the repository meeting its performance objectives regarding criticality. Discuss how the proper function of the engineered barrier system (excluding the waste package) impacts other structures, systems, and components, particularly the waste package and overall postclosure performance regarding criticality.

Present the methods to be used to ensure that subcritical conditions are maintained in the preclosure period under the worst credible conditions. Define the error contingency criteria used to support these methods (NRC 1989a). This information may be presented in Chapter 5 in relation to criticality being incredible in a dry waste package, in which case the discussion in Chapter 6 need only focus on aspects of engineered barrier system (excluding the waste package) design that help ensure the waste package remains dry.

Provide detailed descriptions of the criticality-related engineered barrier system (excluding the waste package) design features. Include dimensions, material descriptions, and manufacturing processes as necessary to support the licensing case. Provide plan and cross-section drawings of the engineered barrier system (excluding the waste package), showing dimensions and tolerances and highlighting criticality-related features as necessary to support the licensing case. Explain why the criticality-related design features were chosen. Describe potential alternative measures, and explain why they were not chosen, as required by 10 CFR 60.21(c)(1)(ii)(D).

6.2.2.4 Engineered Barrier System (Excluding the Waste Package) Structural Design

Provide the structural design criteria for the engineered barrier system (excluding the waste package), as described in the project system description documents.

State the following structural design criteria for the preclosure period (NRC 1996a, Section 7.4.2.2):

- Confinement of radioactive material must be maintained under normal, off-normal, and accident-level conditions. The engineered barrier system (excluding the waste package) must not contribute to release of radioactive material during normal, off-normal, or accident-level conditions.

- The engineered barrier system (excluding the waste package) must not contribute to the deformation of the waste package under credible loading conditions to the extent that the subcritical condition or retrievability of the radioactive material inside the waste package would be jeopardized.

- The engineered barrier system (excluding the waste package) must not contribute to temperatures that would damage the structural integrity of the waste package, the radioactive material inside the waste package, or contribute to thermal degradation of the engineered barrier system (excluding the waste package) and the surrounding underground facility.

- Any radiation shielding provided by the engineered barrier system (excluding the waste package) must not degrade under normal and off-normal conditions due to structural changes or damage to the engineered barrier system (excluding the waste package).

Describe aspects of the design of the engineered barrier system (excluding the waste package) and its components related to structural performance. Discuss any applicable industry codes and standards that were used in the structural design. (Cross-reference the list provided earlier in Section 6.2.2 to avoid repetition.) Include dimensions, material descriptions, and manufacturing processes as necessary to support the licensing case. Provide plan and cross-section drawings of the engineered barrier system (excluding the waste package), showing dimensions and tolerances and highlighting structural-related features as necessary to support the licensing case. Explain why the structural-related design features were chosen. Describe potential alternative measures, and explain why they were not chosen, as required by 10 CFR 60.21(c)(1)(ii)(D).

Concrete Structures

For storage applications, NRC accepts the use of ACI 349-97 (ACI 1997) for the design, material selection, and construction of all reinforced concrete structures important to safety other than concrete reactor vessels. The design, material selection, specification, and construction must also meet any additional or more stringent requirements given in ANSI/ANS-57.9, *Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type)* (ANSI/ANS 1992), as incorporated by Regulatory Guide 3.60, *Design of an Independent Spent Fuel Storage Installation (Dry Storage)* (NRC 1987b). Certain sections of ACI 349-97 (ACI 1997) may be replaced by corresponding sections of ACI 318-95 (ACI 1995), as listed in NUREG-1536 (NRC 1997a), Sections 3.V.2.b.i(2)(b) and 3.V.2.d.iii(2)(a)). For storage applications, NRC accepts the use of either ACI 318-95 (ACI 1995) or ACI 349-97 (ACI 1997) for reinforced concrete structures that are subject to approval but not important to safety (NRC 1997a, Section 3.IV.2 and .3) NUREG-1536 also provides alternative criteria to the temperature requirements of ACI 349-97 (ACI 1997) (NRC 1997a, Section 3.V.b.1(2)(b)). Considering the above information, the project compliance program is assessing the applicability of the documents discussed above (and others). The results of this assessment (CRWMS M&O 1998) are to be discussed in the License Application as follows:

- Regulatory Guide 1.29, *Seismic Design Classification* (NRC 1978a), describes a method acceptable to NRC for identifying and classifying those structures, systems,

and components of a nuclear power plant that should be designed to withstand the effects of the safe shutdown earthquake (i.e., the design basis earthquake). State that, consistent with the guidance in this regulatory guide, hardware in the "ex-container" part of the engineered barrier system (excluding the waste package) identified and classified as required to survive a safe shutdown earthquake will be designed to withstand the effects of a design basis earthquake and remain functional. State that this regulatory guide will be used in conjunction with *Topical Report: Preclosure Seismic Design Methodology for a Geologic Repository at Yucca Mountain* (YMP 1997b) and the (future) topical report on preclosure seismic design bases. Summarize the design criteria that are provided in the topical reports, and reference the topical reports themselves for more detail on the seismic design criteria.

- Regulatory Guide 1.142, *Safety-Related Concrete Structures for Nuclear Power Plants (other than Reactor Vessels and Containments)* (NRC 1981b), provides a methodology acceptable to NRC for complying with the regulations with regard to safety-related concrete structures (other than reactor vessels and containment) for nuclear power plants. State that, consistent with Regulatory Guide 1.142, safety-related concrete structures will be designed to resist a specified magnitude of an earthquake and its aftershocks.

- Regulatory Guide 3.48 (NRC 1989b) describes the standard format and content for the Safety Analysis Report for a commercial independent spent fuel storage installation or DOE monitored retrievable storage installation (dry storage). In accordance with guidance in this document, state in the License Application that the design of engineered barrier system (excluding the waste package) structures will include consideration of :

  - Earthquake, fire, explosion, and differential subsidence effects
  - Normal, off-normal, and special loading and load combinations
  - Allowable foundation loads and deflections and deformation stresses for structures
  - Combination stress loading (Section 4.2)

- Although NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants LWR Edition* (NRC 1987c), is intended to provide guidance for the staff review of nuclear power plants, various sections are general in nature, and are applicable to non-power plant applications. Sections 3.7.1 and 3.8.4 pertain to seismic design parameters and seismic Category I structures, respectively, and can be considered applicable to the waste package support hardware. Section 11.4 provides guidance for solid waste management systems, including selection of material, and will be considered in the fabrication of the ex-container waste package support hardware. State in the License Application that load combinations for service load conditions that use the working stress design method or the strength design method for concrete structures will be in accordance with ACI 318-95, *Building Code Requirements for Reinforced Concrete* (ACI 1995), as discussed in NUREG-0800 (NRC 1987c), Section 3.8.4, p. 3.8.4.-9. Also state that the strength design method

for concrete structures will be used for factored load conditions that represent extreme environmental, abnormal, abnormal/severe environmental, and abnormal/extreme environmental conditions (NRC 1987c, Section 3.8.4, p. 3.8.4-10).

- NUREG-1567, *Standard Review Plan for Spent Fuel Dry Storage Facilities* (NRC 1996a), provides guidance to the NRC staff for the review of applications for licenses for commercial independent spent fuel storage installations and DOE monitored retrievable storage installations. NRC has accepted use of ACI 349-97, *Code Requirements for Nuclear Safety-Related Concrete Structures* (ACI 1997), for design and material selection for reinforced concrete structures important to safety (but not as confinement cask), but has allowed the optional use of ACI 318-95, *Building Code Requirements for Reinforced Concrete* (ACI 1995), for construction. Because NUREG-1567 addresses the design independent spent fuel storage installation or the monitored retrieval storage, which have limited design lives, state in the License Application that the structural guidance from this document is considered to apply to the mined geologic repository structures only up to the end of the preclosure period. State in the License Application that the waste package support hardware structures, systems, and components important to safety will be designed to accommodate the effects of, and to be compatible with, site characteristics and environmental conditions associated with normal operation, maintenance, and testing, and to withstand postulated accidents (Section 3.3, p. 3-2) State that the waste package support hardware design will include the design magnitudes of loads and limits derived from site characteristics and analyses of normal, off-normal, and accident-level conditions. State that such loads include, but are not limited to, dead load, live load, thermal loads, accident loads, earthquake loads, and flood loads (Section 3.4.3.2, p. 3-10). Also, state that live loads for precast structures will consider all loading and restraint conditions from initial fabrication to completion of the structure, including form removal, storage, transportation, and erection (Table 7-1, p. 7-51). Finally, state that reinforced concrete pads that support confinement casks in storage are not "pavements" and will be designed and constructed as foundations under the applicable code, ACI 318-95 (ACI 1995) or ACI 349-97 (ACI 1997) (Section 7.4.4.2, p. 7-31).

- NUREG-1536 (NRC 1997a) provides guidance to NRC for performing safety reviews of dry cask storage systems. Some guidance pertaining to waste canisters and waste storage has been determined to be applicable to the ex-container system. NUREG-1536 contains guidance that is similar to that in NUREG-1567 (NRC 1996a). State the following in the License Application to address this document:

  - Construction of reinforced-concrete structures in the engineered barrier system (excluding the waste package) not important to safety will be in accordance with ACI 318-95 (ACI 1995) (Section 3).

  - Reinforced-concrete pads, which typically are not classified as important to safety, will be designed and constructed as foundations under ACI 318-95 (ACI 1995) or ACI 349-97 (ACI 1997) (Section 3).

- Design, material selection, material specification, and construction for reinforced-concrete engineered barrier system (excluding the waste package) structures important to safety, but not related to confinement (i.e., not within the scope of ACI 359), will be in accordance with ACI 349-97 (ACI 1997), with the optional use of ACI 318-95 (ACI 1995) (instead of ACI 349-97) allowed for construction; additional or more stringent requirements given in ANSI/ANS 57.9 (ANSI/ANS 1992) will also be met (Section 3).

- The design, material selection, material specification, and construction of reinforced-concrete engineered barrier system (excluding the waste package) structures that are not important to safety will be in accordance with either ACI 318-95 (ACI 1995) or ACI 349-97 (ACI 1997). (If ACI 349-97 is used for design, material selection, and material specification, ACI 318-95 may be used for construction (Section 3)).

- Materials and material properties used for the design and construction of reinforced-concrete engineered barrier system (excluding the waste package) structures important to safety, but not related to confinement (i.e., not within the scope of ACI 359, will be in accordance with ACI 349-97 (ACI 1997) (Section 3).

- Materials and material properties used for the design and construction of reinforced-concrete engineered barrier system (excluding the waste package) structures that are not important to safety shall be in accordance with ACI 318-95 (ACI 1995) or ACI 349-97 (ACI 1997) if that code is used for design of the structures (Section 3).

- Embedments for the engineered barrier system (excluding the waste package) concrete components will be in accordance with the code used for design of the reinforced-concrete structure in which they are embedded (i.e., ACI 349-97 (ACI 1997) or ACI 318-95 (ACI 1995)) (Section 3).

- Steel structural attachments to engineered barrier system (excluding the waste package) components will be in accordance with the code used for design of the reinforced-concrete structure in which they are embedded (i.e., ACI 349-97 (ACI 1997) or ACI 318-95 (ACI 1995)) (Section 3).

- Steel structural attachments to engineered barrier system (excluding the waste package) components will be in accordance with American Institute of Steel Construction Specification for Structural Steel Buildings Allowable Stress Design and Plastic Design (published in the American Institute of Steel Construction *Manual of Steel Construction, Allowable Stress Design* (AISC 1991) (Section 3).

- ACI 318-95 (ACI 1995) provides guidance for the proper design and construction of structural members of plain concrete. Among the subjects referenced in this code are permits and drawings; inspections; materials; concrete quality; formwork; control

joints; analysis and design; and structural walls, footing, and pedestals. State that the design and construction of structural concrete members (cast-in-place or precast) will be in accordance with applicable sections of ACI 318-95 (ACI 1995).

- ACI 349-97 (ACI 1997) provides guidance for the proper design and construction of concrete structures that form part of a nuclear power plant and have nuclear safety related functions, but does not cover concrete reactor vessels and concrete containment structures. The structures covered by the code include concrete structures inside and outside the containment system. State that the design and construction of concrete systems, structures, and components important to safety, which support, house, or protect nuclear safety class systems, equipment, or components will be in accordance with applicable sections of ACI 349-97 (ACI 1997).

- ACI 363R-92 (ACI 1992) provides information about high-strength concrete, such as the selection of materials, concrete mix proportioning, batching, mixing, transporting, placing, control procedures, concrete properties, structural design, economics, and applications. State in the License Application that the design and construction of structures with high strength concrete will be in accordance with applicable sections of ACI 363R-92 (ACI 1992).

- ACI 544.1R-82(86) (ACI 1986) provides the state of development of the mechanics for fiber reinforcing of Portland cement concrete by metallic, glass, plastic, and natural fibers along with techniques for mixing and mix proportioning, placing, finishing, and actual and potential applications. State in the License Application that the use and application of fiber reinforced concrete will be in accordance with applicable sections of ACI 544.1R-82(86) (ACI 1986).

- ACI 544.3R-93 (ACI 1993) discusses the process of specifying, mixing, placing, and finishing of steel fiber reinforced concrete. State that the use and application of steel fiber reinforced concrete will be in accordance with applicable sections of ACI 544.3R.93 (ACI 1993).

Include in the License Application the following information for concrete components as needed to support the licensing case (NRC 1997a, Section 3.V.b.ii):

- Dimensioning of all surfaces

- Locations, size, configuration, spacing, welding, enclosure, and depth of cover for reinforcements

- Locations and specifications for control, contraction, and construction joints

- Materials, with defining standards or specifications (cross-reference to materials design subsection as appropriate)

- Information on the physical design of embedments and attachments, including, at a minimum, locations, configuration, depth of embedment, interfaces, material, connections and connectors, and protective or functional coatings

NUREG-1536 (NRC 1997a) states that all concrete used in storage cask systems should be reinforced, regardless of the functional role or need for structural strength and integrity (NRC 1997a, Section 3.V.2.b.i(1)). It may be assumed that a similar expectation will be applied to engineered barrier system (excluding the waste package) concrete components (if any). State that all concrete parts of the engineered barrier system (excluding the waste package) will be reinforced, or describe why any parts will not be (for example, postclosure performance considerations).

Demonstrate that the proposed reinforced concrete design for the engineered barrier system (excluding the waste package) provides levels of ductility required by code (e.g., ACI 349-97 (ACI 1997)). Seismic loads and loads due to other accident conditions or natural phenomena should be shown to be consistent with Appendix C of ACI 349-97 (ACI 1997) (NRC 1997a, Section 3.V. d.i(3)).

Ensure the reinforcing steel quantities, sizes, and locations in reinforced concrete schedules and drawings are consistent with the design analysis in License Application Section 6.2.3. Ensure the potential effects of any excess steel on ductility of the concrete are considered (NRC 1997a, Section 3.V.d.i(3)).

Provide the following design information on concrete engineered barrier system (excluding the waste package) components:

- Upper limit on the specified yield strength of reinforcement and lower limit on concrete specified compressive strength
- Limit on the amount (cross-section area) of compressive reinforcement in flexural members
- Requirements on continuation and development lengths of tensile reinforcement
- Aspects of the design that ensure flexure controls (and limits) the response
- Requirements for shear reinforcement
- Limitations on the amount of tensile steel in the flexural members relative to that which would produce a balanced strain condition
- Projected maximum response to design-basis loads within the permissible ductility ratios for the controlled structural action
- Embedments designed to fail in the steel before pullout from the concrete (NRC 1997a, Section 3.V.d.i(3))

In addition, ensure the construction specifications or descriptions in the Safety Analysis Report demonstrate that substitution of materials, use of larger sizes, or placement of larger quantities of steel than intended in the design will be precluded. Also demonstrate that provisions for splicing or development of reinforcing steel will not reduce ductility of members (NRC 1997a, Section 3.V.d.i(3)).

Describe specifications for planned construction and fabrication that are compliant with the code selected to apply to engineered barrier system (excluding the waste package) concrete structure design. Ensure such specifications prohibit contact between aluminum objects used in construction (for example, chutes, pipes, etc.) and wet concrete. This prohibition must include contact with fine aluminum particles that may be collected by wet concrete in chutes and pipes (NRC 1997a, Section 3.V.b.2.d.iii.(2)(b)).

Demonstrate that the structural design has appropriately considered soil liquefaction or other soil instabilities attributable to vibrating ground motion. NUREG-1536 (NRC 1997a) states that NRC Inspection Procedure 60851 (NRC 1993) and Regulatory Guide 3.60 (NRC 1987b) provide guidance for this analysis (NRC 1997a, Section 3.V.b.i(1)). Describe guidance used.

Non-Concrete Structures

The compliance program (CRWMS M&O 1998) is the source of the following guidance.

For storage applications such as independent spent fuel storage facilities, NRC accepts use of ANSI/ANS-57.9, *Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type)* (ANSI/ANS 1992), together with codes and standards cited within it, as the basic reference for structures important to safety (NRC 1997a, Section 3.IV.4). ANSI/ANS 57.9 (ANSI/ANS 1992) provides design criteria for systems and equipment of an installation for the receipt and dry storage of spent fuel from light water reactors or water pool-type independent spent fuel storage installations. The criteria for the independent spent fuel storage installation design requirements may be applicable to the waste package support hardware. State in the License Application that the engineered barrier system (excluding the waste package) design will be consistent with the requirements provided in Sections 5.4 and 6.4 related to storage areas, including consideration of design loads, materials, and other related issues.

Regulatory Guide 1.29, *Seismic Design Classification* (NRC 1978a), describes a method acceptable to NRC for identifying and classifying those structures, systems, and components of a nuclear power plant that should be designed to withstand the effects of the safe shutdown earthquake, that is, the design basis earthquake. State that waste package support hardware identified and classified as required to survive a safe shutdown earthquake will be designed to withstand the effects of a design basis earthquake and remain functional. State that this regulatory guide will be used in conjunction with *Topical Report: Preclosure Seismic Design Methodology for a Geologic Repository at Yucca Mountain* (YMP 1997b) and the future topical report on seismic design bases. Summarize the design criteria that are provided in the topical reports, and reference the topical reports themselves for more detail on the seismic design criteria.

Although NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants LWR Edition* (NRC 1987c), is intended to provide guidance for the staff review of nuclear power plants, various sections are general in nature, and are applicable to non-power plant applications. Sections 3.7.1 and 3.8.4 pertain to seismic design parameters and seismic Category I structures, respectively, and can be considered applicable to the waste package support hardware. Section 11.4 provides guidance for solid waste management systems, including selection of material, and will be considered in the fabrication of the ex-container waste package support hardware. State, as determined by the compliance program, the extent to which these sections of NUREG-0800 (NRC 1987c) are applied to the design of non-concrete engineered barrier system (excluding the waste package) design.

In NUREG-1536 (NRC 1997a, Sections 3.IV.4 and 3.V.3.b.i(2)), NRC states that the following references may apply to steel structures and components important to safety. The compliance program has addressed some of this guidance, as follows: state in the License Application that embedments for the engineered barrier system (excluding the waste package) concrete components will be in accordance with the code used for design of the reinforced-concrete structure in which they are embedded (i.e., ACI 349-97 (ACI 1997) or ACI 318-95 (ACI 1995)) (Section 3). Also, state that steel structural attachments to engineered barrier system (excluding the waste package) components will be in accordance with the American Institute of Steel Construction Specification for "Structural Steel Buildings Allowable Stress Design and Plastic Design" (AISC 1991) (Section 3).

Some guidance from NUREG-1536 (NRC 1997a) has not yet been addressed by the compliance program. To the extent the below standards are used, describe their use.

- "Specification for Structural Steel Buildings-Allowable Stress Design and Plastic Design," provided in American Institute of Steel Construction *Manual of Steel Construction, Allowable Stress Design* (AISC 1991).

- "Load and Resistance Factor Design Specifications for Structural Steel Buildings," published in the American Institute of Steel Construction *Manual of Steel Construction, Load and Resistance Factor Design* (AISC 1995).

- American Welding Society, *Structural Welding Code Steel*, AWS D1.1-94 (AWS 1993).

- American Society of Civil Engineers, *Minimum Design Loads for Buildings and Other Structures*, ASCE-7-95 (ANSI/ASCE 1996) (the NUREG takes exception that load combinations established on the basis of ANSI/ANS-57.9 (ANSI/ANS 1992) and reflected in NUREG-1536 (NRC 1997a) Table 3-1 are to be used).

For other engineered barrier system (excluding the waste package) structures not important to safety, the following codes and standards listed in NUREG-1536 (NRC 1997a) as potentially applicable to analogous independent spent fuel storage installation equipment may be of use (NRC 1997a, Section 3.V.4.b.i(2)). State in the Safety Analysis Report the extent to which these documents are used:

- ASCE-7-95 (ANSI/ASCE 1996) [Note that this is an updated version of the standard referenced in NUREG-1536 (NRC 1997a)]

- *Uniform Building Code* of the International Conference of Building Officials (ICBO 1994)

- "Specification for Structural Steel Buildings-Allowable Stress Design and Plastic Design," published in the American Institute of Steel Construction *Manual of Steel Construction* (AISC 1995)

- American Institute of Steel Construction *Code of Standard Practice for Steel Buildings and Bridges* (AISC 1992)

Identify the value of design parameters used to meet the design criteria. Describe any uncertainties associated with the parameters and the treatment of those uncertainties. Identify the principal structural members and systems that are important to safety. Reference these items on drawings, and discuss design and performance (NRC 1989a).

Describe the load combinations and factors that serve as engineered barrier system (excluding the waste package) design criteria. For each criterion, state the maximum allowable stresses and strains, as a percentage of yield or ultimate values, for failure. Describe how other structural failure modes are considered in the design. If different criteria are used in various parts of the engineered barrier system (excluding the waste package) for different conditions, indicate the appropriate values for each condition.

Provide mechanical properties of materials used in the structural evaluation, such as yield stress, ultimate stress, modulus of elasticity, ultimate strain, Poisson's ratio, density, and coefficient of thermal expansion.

Identify prequalified welds to be used for the engineered barrier system (excluding the waste package). Provide fabrication drawings and specifications, including dimensions, weld materials, preheat, weld processes, tests, and other commitments that fully define the proposed welds (NRC 1996a, Section 7.4.1.1). Show that welds in low alloy steels will be properly controlled. NUREG-0800 (NRC 1987c) provides four methods for improving quality of welds in low-alloy steel (NRC 1987c, Review Plan 5.2.3). To the extent these methods are used, take credit for them in the License Application. They are:

- Control of cracks by maintaining proper preheat temperatures in the base metal concurrent with controls on other welding variables

- Improvement of electroslag weld quality by maintaining a weld solidification pattern with a strong intergranular bond in the center of the weld

- Adequate accessibility to the weld area

- Limiting occurrence of underclad cracking by avoiding excessive heating during welding (if steels clad with stainless steel are used)

Describe seismic design criteria as they affect the engineered barrier system (excluding the waste package) structural design. (Cross-reference the list provided earlier in Section 6.2.2 to avoid repetition.) Provide a table showing loading combinations (response spectra) and stress criteria to which each component was designed. Reference the seismic topical reports for seismic methodologies used.

For engineered barrier system (excluding the waste package) components that are subjected to seismic or other loadings at elevated temperatures, specify the appropriate mechanical properties under these conditions to the extent they are used in the structural evaluations. Specify the sources for this information, including publication name and page number. If material properties were determined by testing, describe the test procedure, conditions, and measurements in sufficient detail to allow NRC to conclude that the results are valid (NRC 1989a).

State that the engineered barrier system (excluding the waste package) is not subject to tornado and wind loadings and wind-driven missiles because of its sheltered, underground location.

6.2.2.5 Engineered Barrier System (Excluding the Waste Package) Shielding Design

Provide the shielding design criteria for the engineered barrier system (excluding the waste package) using the system description documents as the source of information.

Describe aspects of the engineered barrier system (excluding the waste package) design and its components related to shielding performance and compliance with 10 CFR 60.131(a)(2) and (a)(3). Discuss any applicable industry codes and standards that were used in the shielding design. (Cross-reference the list provided earlier in Section 6.2.2 to avoid repetition.) Identify the value of design parameters used to meet the design criteria. Describe any uncertainties associated with the parameters and the treatment of those uncertainties. Describe considerations of engineered barrier system (excluding the waste package) design that are directed toward ensuring that occupational radiation exposure is as low as reasonably achievable (ALARA). Refer to cost-benefit analysis to support ALARA design. Describe whether experience from past designs was used to develop an improved design for ensuring that incidents of radiation contamination are minimized. Regarding radiation exposure to workers, describe how the engineered barrier system (excluding the waste package) design is directed toward reducing:

- Need for maintenance
- Radiation levels
- Time spent on maintenance (NRC 1989a)

Ensure that drawings, figures, and tables describing shielding features of the engineered barrier system (excluding the waste package) are sufficiently detailed to support in-depth NRC staff evaluation; all information presented in drawings must be legible.

Provide appropriate material properties for shielding materials, and provide a reference for the source of the property values. For nonstandard shielding materials, describe how the properties were validated. State that temperature-sensitive shielding materials will not be subjected to temperatures at or above their design limitations, and refer to analysis in Section 6.2.3 that demonstrates the validity of the statement.

6.2.3 Design Evaluation

The purpose of this section is to describe the evaluation of the compliance of the engineered barrier system (excluding the waste package) and its components with the appropriate design criteria. Describe the interface between the design evaluation described in this section and the repository preclosure radiological safety assessment and postclosure performance assessment described in Chapters 7 and 8, respectively, with particular emphasis on how the engineered barrier system (excluding the waste package) design supports meeting the performance objectives of 10 CFR 60.111 through 10 CFR 60.113, as required by 10 CFR 60.133(h). Provide the projected release rate of any radionuclide from the engineered barrier system (excluding the waste package) following the containment period, and provide a brief summary of the evaluation of that release rate that is contained in Chapter 8 as required by 10 CFR 60.21(c)(1)(ii)(C). This discussion should be coordinated with that in Chapters 5, 7, and 8 to avoid excessive duplication.

State the following and provide evidence to support the assertions:

- All reasonable thermomechanical effects have been considered by DOE, and models used by DOE are unlikely to underestimate the consequences of thermomechanical effects on the structural integrity of the engineered barrier system (excluding the waste package).

- All reasonable environmental effects have been considered by DOE, and models used by DOE are unlikely to underestimate the consequences of thermomechanical effects on the structural integrity of the engineered barrier system (excluding the waste package).

- Models used by DOE are unlikely to underestimate the quantities of gaseous and non-gaseous radionuclides released by the engineered barrier system (excluding the waste package).

- All credible degradation modes of the engineered barrier system (excluding the waste package) have been considered.

- Use of data and conclusions generated in the expert elicitation process for understanding and predicting engineered barrier system (excluding the waste package) behavior is not likely to result in noncompliance with engineered barrier system (excluding the waste package) performance objectives.

Demonstrate that engineered barrier system (excluding the waste package) equipment is being, or has been, designed to have the capability of performing its design functions under all normal, abnormal, accident, and post-accident environments and for the length of time for which the function is required, as demonstrated through appropriate testing and analyses (NRC 1987b, Review Plan 3.11).

Section 8.2.1.2 of Regulatory Guide 4.17, *Standard Format and Content of Site Characterization Plans for High-Level Waste Geologic Repositories* (NRC 1987a), states that waste package support hardware and potential backfill material should be designed with adequate conservatism

with respect to coupled conditions that may be encountered. Demonstrate (if required by the compliance program) that the engineered barrier system (excluding the waste package) design provides this conservatism.

To show compliance with 10 CFR 60.21(c)(1)(ii)(F), describe the modeling methods used to demonstrate that the design parameters are met. Provide an explanation of the measures supporting the models used to perform the analyses. For both design applications and accident analyses, explain measures supporting the models used to perform the analyses. Support analyses and models that have been used to predict future conditions and changes in the system 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. Ensure the following information supporting models is provided in the chapter and/or its references:

- Variability and uncertainty of data and resultant propagation of errors in models or analyses for which such data were used
- Discussions of data representativeness, including uncertainties associated with the extrapolation of data
- Documentation and validation of models and analyses
- Identification of, and justification for, assumptions used in models and analyses
- Input and output data and interpretations of the data, with the basis for the interpretation
- The role of expert judgment, if used, in models and analyses

Provide the following information for models and analyses used to predict postclosure engineered barrier system (excluding the waste package) performance:

- Identification and evaluation of design parameters used to meet design criteria
- Description of uncertainties in parameters and of how these uncertainties are reflected in models
- Descriptions of models and analyses used to predict future conditions and changes in postclosure features of engineered barrier system (excluding the waste package) parameters
- Description of uncertainties in analytical models and how such uncertainties affect predicted results

To demonstrate compliance with 10 CFR 60.21(c)(1)(ii)(E), provide an analysis of the performance of major structures and components of the engineered barrier system (excluding the waste package) to identify those that are important to safety. Describe and analyze the design and performance requirements of these structures and components. Discuss margins of safety

under normal conditions and under conditions that may result from anticipated operational occurrences, including those of natural origin. For purposes of this analysis, assume that geologic repository operations area operations will be carried out at the maximum capacity and rate of receipt of radioactive waste that will be specified in the License Application.

As required by 10 CFR 60.21(c)(6), identify and justify selection of those variables, conditions, or other items that are determined to be probable subjects of license specifications, particularly items that may significantly influence the engineered barrier system (excluding the waste package) design. Coordinate this discussion with the author(s) of Chapter 11, who will be tabulating all such information from all License Application chapters in Section 11.12.

In accordance with 10 CFR 60.21(c)(14), identify structures, systems, and components of the engineered barrier system (excluding the waste package) that require further research and development to confirm design adequacy. Provide a detailed description of the programs designed to resolve safety questions, including a schedule for resolution. Coordinate this discussion with the author(s) of Chapter 1, who will summarize in Section 1.9 all such information from all License Application chapters.

Provide an assessment that describes how the design features of the engineered barrier system (excluding the waste package) comply with containment and release rate requirements. For each of the items, describe how the information presented relates to the design criteria in 10 CFR 60.131, excluding criticality, which is discussed later and in Chapter 5, and the design criteria in 10 CFR 60.133. At a minimum, discuss in detail and supply references for the following:

- Performance assessment codes, including supporting research, testing, and model development (refer to Chapter 8 for discussion of performance assessment codes).
- Assumed anticipated processes and events as well as degradation scenarios.
- Extrapolation of short-term measurements to long-term predictions of engineered barrier system (excluding the waste package) performance (refer to Chapter 8 for discussion).
- Uncertainties in the data, models, codes, and results related to the performance assessments.
- Evaluation of how an engineered barrier system (excluding the waste package) component performs relative to the performance required by the allocation. (Note that the results of analysis are presented in this chapter; the analysis itself is in Chapter 8. Provide appropriate cross-reference.)

Discuss how design criteria for the engineered barrier system (excluding the waste package), including the design criteria from 10 CFR 60.133, have been addressed. Demonstrate how the engineered barrier system (excluding the waste package) designs will not compromise the function of the waste packages, 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.

Show that normal conditions and operations should not result in any degradation of the capabilities of the engineered barrier system (excluding the waste package) during the preclosure period. Show that the engineered barrier system (excluding the waste package) will not contribute to any "wear and tear" of the exterior of the waste package resulting from normal conditions and operations (NRC 1996a, p. 3-9).

If the engineered barrier system (excluding the waste package) will be degraded by design basis events, demonstrate that DOE will develop requirements and procedures for determination and correction of the degradation, or other acceptable remedial action. The description of such plans in the License Application will need to be sufficient to assure NRC that DOE will have the means to correct any engineered barrier system (excluding the waste package) degradation caused by a design basis event (NRC 1996a, Section 7.4.2.1).

As part of the demonstration of compliance for the engineered barrier system (excluding the waste package), list all computer codes used to demonstrate that compliance. Provide the following information for each code to demonstrate its applicability and validity:

- Author, source, dated version, and facility.
- Description of the program, and the extent and limitation of its application.
- Reference to a series of test problems, demonstrating substantial similarity to solutions obtained from any one of the following: hand calculations, analytical results published in literature, experimental tests, a similar program, or benchmark problems. Prepare a summary comparison of the solutions obtained using the selected source, in either graphical or numeric form, and make it available in a reference. Justify any significant deviations between the model and the results from other sources. NUREG-0800 suggests that computer and test solutions should agree within five percent. (NRC 1987c, Review Plan 3.9.1).

In accordance with 10 CFR 60.21(c)(1)(ii)(D), provide a comparative evaluation of alternative engineered barrier system (excluding the waste package) designs and design concepts, with particular emphasis on those features that would provide longer radionuclide containment and isolation. There is no need to show that the design chosen is the best possible design from the standpoint of isolation or from any other perspective. Rather, it is necessary to show a defensible rationale, including safety considerations, practicality, and cost considerations explaining why alternatives were not chosen. The author must show that due diligence was exercised by DOE in investigating potential designs and determining which design best serves the public and national interests. Discussion of alternatives considered with respect to specific design issues, such as criticality or materials, should be provided in the subsections below that contain evaluations of these issues. The discussion in this section takes the results of alternatives evaluations in the subsections that follow and describes the evaluation of alternatives based on all design issues of concern.

NRC Key Technical Issues

The KTIs are defined in a recent NRC annual report (NRC 1997b). Chapter 6 of the License Application should discuss Subissues 2, 3, and 4 of the KTI on Evolution of the Near-Field Environment as described in the paragraphs that follow. Note that this discussion may, at the discretion of the respective chapter authors, occur in chapters other than Chapter 6 (most likely in Chapters 3 and/or 5). Chapter authors are responsible for coordinating discussion of the subissues and acceptance criteria to ensure they are properly and rigorously addressed. Should the decision be made to address the subissues mainly in other chapters, the Chapter 6 text must reference the discussions in the other chapters.

Discuss in general terms how the NRC KTI on Evolution of the Near-Field Environment has been addressed related to the subissues discussed below. Include a discussion of how NRC sees the issue as discussed in its related recent IRSR, and provide the DOE perspective on the status of resolution of the issue/subissue/acceptance criteria at the time of preparation of the License Application. Address each of the acceptance criteria in these subissues in a manner that demonstrates resolution of each criterion. Any aspect of any criteria not resolved should be the subject of plans to obtain additional information and should be included in the description of such activities in Section 11.11 of the technical guidance document.

KTI: Evolution of the Near-Field Environment

The acceptance criteria text that follows is a direct quote from the most recent IRSR on this KTI (NRC 1997c). The discussion provides the NRC perspective on what DOE should have provided to them to resolve the related subissue. (See the technical guidance document introduction for discussion regarding why some acceptance criteria from the referenced document may not be listed below.)

Subissue 2: The Effects of Coupled Processes on the Waste Package Lifetime

Acceptance Criteria:

DOE identified waste package lifetime (containment) as a key factor that is important to its waste isolation strategy. Important factors affecting containment are:

(i) relative humidity;

(ii) seepage and the presence or absence of liquid water on the waste package;

(iii) waste package degradation mechanisms, such as general corrosion, pitting corrosion, and microbial attack;

(iv) protection of the inner barrier provided by the outer barrier of a double-walled container;

(v) cathodic protection of the inner barrier;

(vi) waste package temperature;

(vii) thermal heat load of a waste container;

(viii) radiolysis effects on corrosion (e.g., production of nitric acid);

(ix) deposition of salts on the waste container; and

(x) performance of backfill; and others.

Waste package containment is embodied in the NRC performance assessment subsystem abstraction element of waste package corrosion. DOE must adequately estimate the waste package lifetime and appropriately consider processes affecting containment in its assessments of waste containment and isolation.

DOE included the following relevant processes and any others that may affect waste package lifetime in its analyses:

(i) thermal-hydrologic effects on waste package lifetime;

(ii) hydrothermal-chemical effects, such as changes in groundwater chemistry that may affect waste package lifetime;

(v) effects of microbial processes on waste package lifetime.

Data and models have been collected, developed, and documented under acceptable quality assurance procedures.

Subissue 3: The Effects of Coupled Processes on the Rate of Release of Radionuclides from Breached Waste Packages

Acceptance Criteria:

DOE has identified radionuclide mobilization from the waste form as a key factor affecting dose. Radionuclide release rates and solubility limits constitute one of the NRC key elements of subsystem abstraction for performance assessment. DOE must adequately estimate rates of radionuclide release and appropriately consider these rates and processes affecting them in its assessments of waste containment and isolation.

DOE is to include the following processes and any others that may affect rate of release in its analyses:

(i) thermal-hydrologic effects on liquid flow;

(ii) hydrothermal-chemical effects, such as changes in groundwater chemistry that may affect rate of release;

(v) effects of cementitious materials on chemical conditions and hydraulic properties affecting rate of release.

Data and models have been collected, developed, and documented under acceptable quality assurance procedures.

NRC staff should verify that there are no deficiency reports concerning data quality related to effects of coupled processes on the rate of radionuclide release that have not been closed.

Subissue 4: The Effects of Coupled Processes on Radionuclide Transport Through Engineered and Natural Barriers

Acceptance Criteria:

DOE considers radionuclide transport a key performance attribute of the proposed repository. Retardation of radionuclides in fractures in the unsaturated zone and in the saturated zone constitute two of the NRC key elements of subsystem abstraction for performance assessment. DOE must adequately estimate the radionuclide transport characteristics of the near field and appropriately consider radionuclide transport in its assessments of waste containment and isolation.

Data and models have been collected, developed, and documented under acceptable quality assurance procedures.

NRC staff should verify that there are no deficiency reports concerning data quality in relation to radionuclide transport that have not been closed.

6.2.3.1 Engineered Barrier System (Excluding the Waste Package) Materials Performance

Describe in this section the evaluation of the compliance of the engineered barrier system (excluding the waste package) materials and its components with the approved design criteria. Describe the interface between the design evaluation described in this section and the overall repository preclosure and postclosure performance assessment described in Chapters 7 and 8.

Demonstrate the acceptability of corrosion resistance of engineered barrier system (excluding the waste package) materials under prospective environments. Cite controls to be placed on material quality, or state that appropriate quality controls are included in codes that are incorporated by reference in the design, fabrication, and construction criteria (NRC 1997a, Section 3.V. 3.c).

As required by 10 CFR 60.21(c)(1)(ii)(F), provide an explanation of measures supporting the models used to perform analyses for both design applications and accident 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 tests, in situ tests, laboratory tests that are representative of field conditions, monitoring data, and natural analog studies.

Discuss possible chemical, galvanic, or other reactions between the waste package and the engineered barrier system (excluding the waste package). Describe any measures to be taken to

prevent contact or reaction between materials and discuss the predicted or known effectiveness of such measures (NRC 1989a).

If DOE proposes to use a new engineered barrier system (excluding the waste package) material or construction technique not addressed in prior NRC approvals, ensure sufficient test and other data is provided to establish, for the NRC reviewer, the acceptability of the material.

In accordance with 10 CFR 60.21(c)(1)(ii)(D), provide a comparative evaluation of alternative engineered barrier system (excluding the waste package) materials. There is no need to show that the materials chosen are the best possible from the standpoint of isolation or from any other perspective. Rather, it is necessary to show a defensible rationale, including safety considerations, practicality, and cost considerations, explaining why the alternatives were not chosen. In other words, the author must show that due diligence was exercised by DOE in investigating potential designs and determining which materials design best serves the public and national interests. Materials considerations are a subset of all considerations that factor into selection of the engineered barrier system (excluding the waste package) design. The discussion of alternatives in this subsection should refer to the discussion in Section 6.2.2 that describes the overall assessment of alternative designs for the engineered barrier system (excluding the waste package) based on consideration of all appropriate factors.

6.2.3.2 Engineered Barrier System (Excluding the Waste Package) Thermal Performance

Describe the evaluation of the compliance of the engineered barrier system (excluding the waste package) and its components with the thermal design criteria. Provide a cross-reference to Sections 3.7 and 4.4 of the License Application. Describe the interface between the design evaluation described in this section and the overall repository preclosure and postclosure performance assessment described in Chapters 7 and 8.

As required by 10 CFR 60.21(c)(1)(ii)(F), describe the thermal modeling methods used to demonstrate that the design parameters are met. Include data on individual components and the engineered barrier system (excluding the waste package) as a whole. Fully justify modeling assumptions and demonstrate that the thermal model is conservative. Provide an explanation of the measures supporting the models used to perform the analyses. Provide support for analyses and models used to predict future conditions and changes in the engineered barrier system (excluding the waste package) thermal conditions. Provide an explanation of measures supporting the models for both design applications and accident analyses, an explanation of measures supporting the models used to perform analyses (support analyses and models that have been used to predict future conditions and changes in the system 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).

Describe the analytical thermal model in detail. Ensure the dimensions and materials shown in the sketches or figures of the model are consistent with the drawings of the same items presented elsewhere in this Safety Analysis Report chapter and with model computer inputs. Discuss treatment of gaps between components; show that tolerances are considered so that the thermal resistance of each gap is treated conservatively. For engineered barrier system (excluding the

waste package) components important to heat removal, describe and justify treatment in the model of manufacturing techniques for joining components, surface roughness, contact pressure, and gap conductance values. Ensure the heat load that is an input to the model is consistent with the heat load profile of the waste package.

Discuss the variability and uncertainty of data and the propagation of errors. Include evaluations of data representativeness, as well as uncertainties associated with the extrapolation of data. Discuss conceptualizations and the documentation and validation of codes and models used with respect to uncertainties related to the data on which the model is based, the applicability of specific models, the appropriateness of assumptions used in modeling, and the sensitivity of model results to the uncertainty of the input data. Provide input and output data and interpretations along with the basis for the interpretation. Document the role of expert judgment when it was used.

Describe the computer codes used for thermal evaluation and show that they have been well verified and validated. Show that the number of dimensions and temporal treatment are appropriate for the calculations being performed. Provide input and output file listings for thermal evaluations (NRC 1997a, Section 4.V.5.a).

Use of effective thermal conductivity coefficients for gaps may overestimate heat transfer (NRC 1997a, Section 4.V.4.a). If effective thermal conductivity is used, use test data if possible to support the values selected. Also, provide support for values of surface emissivities selected for radiation heat transfer.

Regulatory Guide 3.54 (NRC 1984) provides guidance for the calculation of spent fuel heat generation in an independent spent fuel storage installation. The calculation applicable to the waste package support hardware is an input to the design of the support hardware and for the possible use of backfill material. State that values of heat generation rates for use as design input for the installation will be calculated in accordance with Regulatory Guide 3.54 as applicable (Section C). This document is being revised (issued as DG-3010, September 1997, for comment). The proposed revision has no impact on the statement to be placed in the License Application.

Provide a cross-reference to Sections 3.7 and 4.4 of the License Application for a description of the tests, models, and procedures used to correlate the test data to the thermal environment for normal conditions. Document the effect of the engineered barrier system (excluding the waste package) on the maximum temperature distribution for the waste package versus time, including the spent fuel and major waste package components. Identify the normal conditions that result in the worst combinations of thermal loadings to the waste package and to the engineered barrier system (excluding the waste package) and surrounding host rock environment. Consider the effects of phase change, gas generation, and chemical decomposition (NRC 1989a). Identify and describe the major mechanisms by which the engineered barrier system (excluding the waste package) transfers heat. Ensure the discussion and assumptions here are consistent with the waste package thermal analysis presented in License Application Chapter 5.

Present the results of evaluation of engineered barrier system (excluding the waste package) thermal performance, including overall performance and component performance, for normal

and off-normal conditions. Take into account significant conditions to be found in the ranges bounded by maximum and minimum drift temperatures and minimum and maximum decay heat loads over the period of regulatory concern. Compare the results with allowable limits on factors such as pressure and temperature for the waste package components and limits on engineered barrier system (excluding the waste package) materials (NRC 1989a).

Ensure thermal parameter inputs used in analysis of engineered barrier performance are consistent with analogous inputs elsewhere in the License Application (for example, spent fuel heat rate). Discuss the variation in thermal properties (for example, thermal conductivity, heat capacity, and density) of engineered barrier system (excluding the waste package) components with temperature changes and with time (that is, with degradation of the components being analyzed); also consider anisotropic dependencies. Demonstrate that these changes support assumptions made in the thermal analysis regarding maximum temperatures and temperature gradients. Thermal properties should be traced to an authoritative reference (generally not a textbook). NRC has accepted the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code*, Division 1, Section II, "Material Specifications" (ASME 1992) and Section III appendices (ASME 1989) as a primary source for material properties. Take credit in the License Application for use of this standard if it is used. If regional thermal properties are determined from a combination of individual materials, describe the manner in which these properties were calculated (NRC 1997a, Section 4.V.4.b).

Demonstrate that the thermal analysis as applied to the engineered barrier system (excluding the waste package) appropriately accounts for thermal interactions among waste packages arrayed in a drift.

Ferritic materials are subject to brittle fracture at low temperatures. Discuss the susceptibility of engineered barrier system (excluding the waste package) structures and components to this phenomenon.

Discuss the analyses or tests to verify that the engineered barrier system (excluding the waste package) will perform, within some defined variance, in accordance with the results of the thermal analyses or tests for normal conditions of storage. Describe the analysis or test setup. Include in the description heat sources, instrumentation, and schematics showing thermocouple and heat source locations as well as the placement of other test equipment. Estimate test sensitivities based on instrumentation, test item, and environmental variations. Discuss the procedures used in all tests and describe the data-recording method. Report the frequency of data recording during the test. Discuss the criteria used to define the steady-state (thermal equilibrium) condition of the test item. Discuss the thermal acceptance criteria and the method employed to compare any acceptance test results with predicted thermal performance. Discuss the action to be taken if the thermal acceptance criteria are not met (NRC 1989a).

Describe the performance of the engineered barrier system (excluding the waste package) in conditions other than normal. Determine and analyze maximum thermal gradients, considering changes in temperature over time. Include a discussion of the ability of the engineered barrier system (excluding the waste package) to function effectively from a thermal standpoint should the engineered barrier system (excluding the waste package) be covered with debris as a result of rockfalls (NRC 1997a, Section 4.V.2.b.(4)(d)). Demonstrate, at a minimum, that the following

scenarios do not cause engineered barrier system (excluding the waste package) thermal limits to be exceeded:

- Maximum normal, off-normal, and accident-level ambient temperatures
- Retrieval operations or preparations for such operations (that is, during forced cooldown of the drift and the structures, systems, and components in the drift)
- Fire or explosion

For fires, specify the assumed duration and flame temperature, as well as the flame velocity and emissivity. NUREG-1536 (NRC 1997a) accepts use of flame and cask surface emissivities specified in 10 CFR 71.73(a)(3) for fires involving a storage cask; should this guidance be used in the fire analysis, so state. If not, state the source of values used. Identify and describe mechanisms and models for coupling the fire energy to the cask surface. These mechanisms include forced convection in relation to flame velocity, as well as radiation. Justify convection coefficients assumed for the fire analysis. Natural convection coefficients are not appropriate, as they imply downward gas flow adjacent to relatively cool component surfaces. In general, buoyant upward flow will dominate (NRC 1997a, Section 4.V.4.c).

NRC accepts that concrete temperatures may exceed the temperature limits of ACI 349-97 (ACI 1997) for fire accidents involving storage casks. If analysis shows these limits to be exceeded for concrete engineered barrier system (excluding the waste package) components (if any), state what corrective action would be performed. NRC also accepts simplifying assumptions for the effects of reinforcing steel in determining the thermal performance and temperature distributions of reinforced concrete. Use of a homogeneous material, instead of modeling the concrete and steel as separate elements, is acceptable if the substitute hypothetical material has appropriately adjusted thermal properties, and the reinforcing steel is covered with concrete in accordance with code. More specific analyses may be required for thermal performance and/or temperature distributions of reinforced concrete designs with features that allow significant thermal transfer below the concrete surface (NRC 1997a, Section 4.V.5.d).

Provide a comparative evaluation of alternative engineered barrier system (excluding the waste package) designs and design concepts as required by 10 CFR 60.21(c)(1)(ii)(D), with regard to design features that would be beneficial from a thermal standpoint. There is no need to show that the design chosen is the best possible from the standpoint of isolation or from any other perspective. Rather, it is necessary to show a defensible rationale, including safety considerations, practicality, and cost considerations, explaining why the alternatives were not chosen. In other words, the author must show that due diligence was exercised by DOE in investigating potential designs and determining which thermal design features best serve the public and national interests. Thermal considerations are a subset of all conditions that factor into selection of the engineered barrier system (excluding the waste package) design. The discussion of alternatives in this subsection should refer to the discussion that describes the overall assessment of alternative designs for the engineered barrier system (excluding the waste package) based on consideration of all appropriate factors.

6.2.3.3 Engineered Barrier System (Excluding the Waste Package) Criticality Performance

Note that criticality analysis is primarily discussed in Chapter 5 of the License Application. The Chapters 5 and 6 authors should work together to minimize duplication in discussion of criticality analysis. It may be appropriate to put little information on criticality evaluation in this section of Chapter 6, referring the reader to Chapter 5 for the detailed information and references. On the other hand, the authors may elect to discuss internal criticality (that is, criticality inside the waste package) in Chapter 5 and external criticality (criticality outside the waste package) in Chapter 6. Criticality analysis will need to include evaluation of potential external criticalities beyond as well as in the engineered barrier system (excluding the waste package). Therefore, this technical guidance document has been developed based on the assumption that the entire criticality evaluation will occur in Chapter 5, with primarily cross-references in Chapter 6.

The following information is written assuming a role exists for engineered barrier system components beyond those comprising the waste package. However, if the contribution of those engineered components, excluding the waste package, to criticality control is not germane and/or no credit is being taken in criticality analyses for engineered components other than those considered part of the waste package, then this needs to be stated. Provide references to Chapter 5 to address criticality issues.

Discuss the effect, if any, of the engineered barrier system (excluding the waste package) components on preclosure and postclosure criticality. Include lists of input assumptions and lists of computer codes used in the analyses. Present the criteria for verifying models or computer programs used in criticality analyses (NRC 1989a). Since most of the criticality issues will be discussed in Chapter 5, it is only necessary to provide discussions of how the following requested information is impacted by the engineered barrier system (excluding the waste package).

Describe an evaluation of the compliance of the engineered barrier system (excluding the waste package) and its components with the appropriate design criteria related to criticality control. Describe the interfaces among the design evaluation described in this section, the waste package design evaluation in Chapter 5, and the overall repository preclosure and postclosure performance assessments described in Chapters 7 and 8.

As required by 10 CFR 60.21(c)(1)(ii)(F), describe the modeling methods used to demonstrate that the design parameters are met. Provide an explanation of the measures supporting the models used to perform the analyses. Provide support for analyses and models used to predict future conditions and changes in the waste package criticality environment, including the engineered barrier system (excluding the waste package) environment. Provide an explanation of measures supporting the models. Provide, for both design applications and accident analyses, an explanation of measures supporting the models used to perform analyses. Support analyses and models that have been used to predict future conditions and changes in the system 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.

Discuss the variability and uncertainty of data and the propagation of errors. Include evaluations of data representativeness, as well as uncertainties associated with the extrapolation of data. Discuss conceptualizations and the documentation and validation of codes and models used with respect to uncertainties related to the data on which the model is based, the applicability of specific models, the appropriateness of assumptions used in modeling, and the sensitivity of model results to the uncertainty of the input data. Provide input and output data and interpretations along with the basis for the interpretation. Document the role of expert judgment when it was used.

Refer to Chapter 5 to provide a description of the method used to calculate the effective multiplication factor of the waste package under normal conditions of storage and accident conditions. Refer to Chapter 5 for a description of the computer program and neutron cross sections used with their referenced documentation. Discuss the basis for selecting the program and cross sections (NRC 1989a).

If an experimental method was used to determine the compliance of the engineered barrier system (excluding the waste package) and waste package with criticality requirements, include a complete description of the method and a discussion demonstrating that the method conservatively takes into account both normal and accident conditions of storage for the waste package (NRC 1989a).

Assuming emplacement of the waste package and with the engineered barrier system (excluding the waste package) in place, reference Chapter 5 for a demonstration that the maximum reactivity for fuel loading or other contents loading has been evaluated for both a single waste package and arrays of waste packages for normal and accident conditions. Itemize and discuss approximations, boundary conditions, calculational convergence criteria, and cross-section adjustments (NRC 1989a).

Refer to Chapter 5 to provide the results of the reactivity calculations establishing the most reactive configurations, with the engineered barrier system (excluding the waste package) in place, for a single waste package and arrays of waste packages for both normal conditions of storage and accident conditions. Justify any interpolations and extrapolations. Discuss the validity and conservatism of the analysis, including the bias established with the benchmark calculations (NRC 1989a). Refer to Chapter 5 for details regarding calculation.

Justify and show the validity of the calculational method and neutron cross-section values used in the analyses. Regulatory Guide 3.4, *Nuclear Criticality Safety in Operations with Fissionable Materials at Fuels and Materials Facilities* (NRC 1986b), provides information on validation of criticality calculations. Refer to Chapter 5 of this technical guidance document for details on how to discuss use of this regulatory guide.

Provide a general discussion of selected critical benchmark experiments that are to be analyzed using the method and cross sections given in Chapter 5. Show the applicability of the benchmarks in relation to the engineered barrier system (excluding the waste package) design. Provide references giving documentation on these benchmarks (NRC 1989a). Provide the results of the benchmark calculations. Establish and provide a discussion of any calculation bias (NRC 1989a).

As required by 10 CFR 60.21(c)(1)(ii)(D), provide a comparative evaluation of alternative engineered barrier system (excluding the waste package) designs and design concepts, with regard to design features that would be beneficial from a criticality control standpoint. There is no need to show that the design chosen is the best possible from the standpoint of isolation or from any other perspective. Rather, it is necessary to show a defensible rationale, including safety considerations, practicality, and cost considerations, explaining why the alternatives were not chosen. In other words, the author must show that due diligence was exercised by DOE in investigating potential designs and determining which criticality-related design features best serve the public and national interests. Criticality considerations are a subset of all conditions that factor into selection of the engineered barrier system (excluding the waste package) design. The discussion of alternatives in this section should refer to the discussion that describes the overall assessment of alternative designs for the engineered barrier system (excluding the waste package) based on consideration of all appropriate factors.

6.2.3.3.1 Preclosure Criticality

For the preclosure period, the rule with which compliance must be demonstrated is 10 CFR 60.131(h). This rule requires that criticality not be possible without two unlikely, independent, and concurrent or sequential changes in conditions necessary for criticality. It also requires accounting for uncertainties in the analysis method and imposes a five percent administrative margin to help ensure criticality does not occur.

State at the beginning of the subsection that DOE will design the engineered barrier system and waste package and will operate the repository such that a criticality event in the underground facility during the preclosure period will not occur absent the unlikely, independent conditions specified in 10 CFR 60.131(h). Refer to Chapter 5 for discussion of criticality inside a loaded and sealed waste package that address the conditions in 10 CFR 60.131(h). Chapter 5 provides analyses that support this position.

State that multi-package criticality is not possible with intact waste packages, even at the closest possible waste package spacing and greatest possible nuclear reactivity of adjacent packages. Support the statement with reference to analysis in Chapter 5. Discuss the waste package and engineered barrier system emplacement controls, if required, provided to ensure multi-package criticality does not occur. Having shown that without water inside the waste package, criticality will not occur except as specified in 10 CFR 60.131(h), discuss the contribution, if any, of the engineered barrier system (excluding the waste package) to ensure that waste package penetration after emplacement will not occur in the preclosure period.

6.2.3.3.2 Postclosure Criticality

No postclosure criticality rule exists other than 10 CFR 60.131(h). Changes to this regulation (or a new regulation) to more clearly address postclosure criticality are likely.

DOE plans to submit a topical report on the subject of postclosure criticality analysis. Should this report be accepted by NRC prior to submittal of the License Application, the DOE compliance case will largely consist of application of the accepted methodology. Should the

topical report not yet have been accepted by the submittal date of the License Application, it may be referred to similar to any other reference, recognizing NRC will need to review the reference.

Divide the postclosure criticality demonstration into three parts:

- Criticality inside the waste package (Chapter 5)
- Criticality in the near field (Chapter 5 or 6 at the author's discretion)
- Criticality in the far field (Chapter 5 or 6 at the author's discretion)

For each case, demonstrate compliance of the engineered barrier system (excluding the waste package) with 10 CFR 60.131(h) or its successor postclosure criticality regulation. Refer to Chapter 5 for presentation of the criticality analysis, assuming the authors choose to present the entire discussion there.

Several NRC regulatory guides, American Nuclear Society standards, and American National Standards Institute standards regarding criticality are available. The following regulatory guides and standards may have potential applicability to the engineered barrier system (excluding the waste package). The project will assess the applicability of these documents to the project in general and to the engineered barrier system (excluding the waste package) specifically and will provide appropriate instruction for the License Application author in a future revision of this technical guidance document. See Chapter 5 of this technical guidance document for information on how to discuss these documents in the License Application.

In Regulatory Guide 3.4, *Nuclear Criticality Safety in Operations with Fissionable Materials at Fuels and Materials Facilities* (NRC 1986b), NRC has endorsed use of ANSI/ANS-8.1-1983, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors* (ANSI/ANS 1983a), for general storage and transport of fissionable materials. This standard provides guidance for the prevention of criticality accidents in the handling, storing, processing, and transporting of certain fissionable material, specifically U-233, U-235, and Pu-239. It provides basic criteria and limits for certain simple geometries of fissionable materials. It also states requirements for establishing validity and ranges of applicability of any calculational method used in assessing criticality safety.

Regulatory Guide 3.58, *Criticality Safety for Handling, Storing, and Transporting LWR Fuel at Fuels and Materials Facilities* (NRC 1986a), endorses ANSI/ANS-8.17-1984 (R1989), *Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors* (ANSI/ANS 1989), for storage and transportation of light water reactor spent fuel, though it takes exception to verification of exposure history as an acceptable method to verify burnup credit.

ANSI/ANS-8.15-1981, *American National Standard for Nuclear Criticality Control of Special Actinide Elements* (ANSI/ANS 1981), provides guidance for prevention of criticality accidents in the handling, storage, processing, and transportation of special actinide elements. The document provides guidance for 14 nuclides ranging from Np-237 to Cf-251. This standard is the counterpart of ANSI/ANS-8.1-1983 (ANSI/ANS 1983a) for materials that, while generally much less abundant than those within the scope of ANSI/ANS-8.1-1983, are nevertheless a potential criticality concern.

ANSI/ANS-8.10-1983, *Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement* (ANSI/ANS 1983b), though intended for application to fissionable material process facilities outside of reactors, could be interpreted to apply to the postclosure repository, in which adequate protection, including shielding provided by the rock surrounding the repository, for the public against radiation and release of radioactive materials can be demonstrated. The approach described in ANSI/ANS-8.10-1983 (ANSI/ANS 1983b) requires designing for one unlikely event rather than for two unlikely events as required by ANSI/ANS-8.1-1983 (ANSI/ANSI 1983a) and ANSI/ANS-8.17-1984 (R1989) (ANSI/ANS 1989).

6.2.3.4 Engineered Barrier System (Excluding the Waste Package) Structural Analysis

In this section, describe the evaluation of the compliance of the engineered barrier system (excluding the waste package) and its components with the structural design criteria. Describe the interface between the design evaluation described in this section and the overall repository preclosure and postclosure performance assessment described in Chapters 7 and 8. Describe the modeling methods used to demonstrate that the design parameters are met. Provide an explanation of the measures supporting the models used to perform the analyses. Provide support for analyses and models used to predict future conditions and changes in the engineered barrier system (excluding the waste package) environment.

It is important that this section show NRC reviewers that the DOE design and analysis procedures and assumptions are conservatively defined on the basis of accepted engineering practice. If DOE proposes to exceed accepted limits for certain load combinations at localized points on the structure, ensure adequate justification is provided to demonstrate that the deviation will not affect the functional integrity of the structure (NRC 1997a, Section 3.V).

In NUREG-1536 (NRC 1997a), NRC requires inclusion of several normal and off-normal conditions that may be of concern for reinforced concrete structures (NRC 1997a, Section 3.d.i) by analogy, the following conditions should be analyzed and results provided in this section as applicable:

- Live and dynamic loads associated with transfer of the waste package to its emplacement position
- Load or support conditions associated with potential differential settlement of foundations
- Thermal gradients associated with the normal range of operations
- Thermal gradients that may result from impingement of water on highly heated concrete
- Live and dynamic loads associated with equipment or instrument malfunctions, or accidental misuse during transfer of the waste package to its emplacement position

- Situations in which the waste package or its handling system jam against portions of the engineered barrier system (excluding the waste package) already installed at the time the waste package is emplaced

- Potential impact on engineered barrier system (excluding the waste package) reinforced concrete components of a suspended waste package (if applicable)

- Loads associated with accidental drops or other impacts during transfer or emplacement of waste packages

- Accidents or events that produce extreme thermal gradients in concrete

- Contact between the waste package and the engineered barrier system (excluding the waste package) concrete components caused by earthquakes

Unless the lowest margins of safety in reinforced concrete structures have been determined by finite-element analysis using the applicable load combinations, identify critical sections for each structurally distinct element of the concrete structure. NUREG-1536 (NRC 1997a) states that the level of refinement in identifying critical sections depends primarily on the margins of safety and secondarily on the importance of the structure to safety. The document goes on to state that structures important to safety may have such high margins of safety that only elementary structural computations are necessary to demonstrate compliance with all the applicable load combinations. For simple elementary analysis, NUREG-1536 (NRC 1997a) states that the margin of safety for a particular section should consider the highest axial, bending, and shear stresses occurring concurrently. On the other hand, intensive analysis is expected to be needed to prove that truly critical sections are used when margins of safety are near the minimum acceptable values. Therefore, justify the approach used to identify critical sections (NRC 1997a, Section 3.V.2.d.iii(1)).

NUREG-1536 (NRC 1997a) also states that NRC does not require analysis of load combinations for situations in which nuclear material is not present. In consideration of the repository situation, this would mean that such analysis would not be necessary for installed parts of the engineered barrier before the waste package is emplaced in their vicinity. However, NUREG-1536 (NRC 1997a) also requires that reinforced concrete structures not be exposed to credible damage that may not be discovered or evident before completion of construction or use. Describe the controls that will prevent undetected damage to the engineered barrier system (excluding the waste package) prior to waste package emplacement (NRC 1997a, Section 3.V.2.d.iii(1)).

Demonstrate through analysis of structural responses to accidents that the concrete engineered barrier system (excluding the waste package) structures, systems, and components important to safety can continue to perform their safety functions. NUREG-1536 (NRC 1997a) states that design codes and load combinations for such equipment can permit permanent damage or deformation under design basis event loading (NRC 1997a, Section 3.V.2.d.iii(3)). If analysis shows that structural damage could occur, demonstrate continuing capability with regard to essential functional performance (including supporting retrievability). Describe consideration of

the nature and extent of credible damage and potential interactions between the damaged reinforced concrete structure and other structures, systems, and components important to safety.

NUREG-1536 (NRC 1997a) states that reinforced concrete structures are not required to survive an accident event with the same capability for a full design life and the same ability to withstand further accidents. Degradation should be readily apparent in the course of routine inspections and surveillances (NRC 1997a, Section 3.V.2.d.iii(3)). State that damage from preclosure accidents is expected to be discovered by post-accident tests or inspections, and that analyses will be performed post-accident to determine the extent of damage, to be followed by repairs or equipment replacements as needed in the preclosure period to ensure the affected components will perform as designed in the preclosure and postclosure periods.

NUREG-0800 (NRC 1987c, Review Plan 3.5.3) states that sufficient concrete thickness must be provided to prevent perforation, spalling, or scabbing of concrete barriers in the event of missile impact. Show (if required by the compliance program) that this criterion is met for any concrete barrier materials in the engineered barrier system (excluding the waste package).

Demonstrate through analysis that engineered barrier system (excluding the waste package) structures not important to safety or waste isolation can properly perform their intended functions and that the response of these structures to credible off-normal and accident conditions will not create secondary hazards for waste packages, the waste inside them, or other structures, systems, or components that are important to safety or to waste isolation.

As required by 10 CFR 60.21(c)(1)(ii)(F), for both design applications and accident analyses, provide an explanation of measures supporting the models used to perform analyses. Support analyses and models that have been used to predict future conditions and changes in the system 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.

NUREG-1536 (NRC 1997a) states that NRC has accepted use of different analytical codes and models for structural analysis of reinforced concrete structures. It further states that NRC does not require use of computer models and codes for analysis of responses or stresses of simple spent fuel storage concrete structures, nor does NRC require that the codes used have been developed under rigid nuclear safety quality controls. However, it states that codes must be appropriately validated for their intended use. For complex reinforced concrete structures, NRC has accepted use of analytical codes intended for dynamic analysis. The basis for determining the acceptability of the DOE approach will be that further refinement of the approach would be expected to have negligible effect on the conclusion. Describe the validation process for structural codes used. Show why codes selected for given applications are sufficiently refined (NRC 1997a, Section 3.V.2.d.ii(3)).

Discuss the variability and uncertainty of data and the propagation of errors. Include evaluations of data representativeness, as well as uncertainties associated with the extrapolation of data. Discuss conceptualizations and the documentation and validation of codes and models used with respect to uncertainties related to the data on which the model is based, the applicability of specific models, the appropriateness of assumptions used in modeling, and the sensitivity of model results to the uncertainty of the input data. Input and output data and interpretations also

should be provided along with the basis for the interpretation. Document the role of expert judgment if used.

For each design basis event that could affect the structural integrity of the engineered barrier system (excluding the waste package), describe the course of the event and describe analyses that show the structural consequences for the engineered barrier system (excluding the waste package). Demonstrate that the design criteria have been met. If the life of the engineered barrier system (excluding the waste package) may be degraded by design basis events, state in the design evaluation that DOE will develop requirements and procedures for determination and correction of the degradation, or other acceptable remedial action. The description of such plans in the License Application will need to be sufficient to assure NRC that DOE will have the means to correct any engineered barrier system (excluding the waste package) degradation caused by a design basis event (NRC 1996a, p. 7-12).

Discuss compliance of the engineered barrier system (excluding the waste package) structural components with 10 CFR 60.133(a)(2). State that because those components of the facility constituting the engineered barrier system (excluding the waste package) discussion in Chapter 6 are underground, tornado and wind loading is not an issue. If flooding of the underground is a credible event, demonstrate that the probable maximum flood will not cause unacceptable damage to the engineered barrier system (excluding the waste package) or its components. Discuss effects of flood-borne debris, if credible. Show that methods of calculating loadings on structures, systems, and components induced by flooding provide acceptably conservative results (NRC 1987b, Review Plan 3.4.2).

For seismic events, identify the methods of seismic analysis used for all engineered barrier system (excluding the waste package) components. Reference the DOE *Topical Report: Preclosure Seismic Design Methodology for a Geologic Repository at Yucca Mountain* (YMP 1997b).

For combined loads, describe the criteria selected to provide mechanical and structural integrity. Define loads and loading combinations to which the engineered barrier system (excluding the waste package) is designed, including load factors selected for each load component in which a factored load approach is used. Specify the design approach used with the loading combination. Provide design loading combinations used to examine the effects on localized areas, such as penetrations, structural discontinuities, and local areas of high thermal gradients, together with time-dependent loading, thermal effects, effects of creep and shrinkage, and other related effects (NRC 1989a).

Summarize the pressures and temperatures, determined in the thermal evaluation in Section 6.2.3.2, that the engineered barrier system (excluding the waste package) and its components will experience in the preclosure and postclosure periods. Calculate deformations and stresses, if any, in engineered barrier system (excluding the waste package) components that result from differential thermal expansion in the waste package. Consider steady-state and transient conditions. These calculations must be sufficiently comprehensive to demonstrate engineered barrier system (excluding the waste package) and waste package integrity under normal conditions (NRC 1989a).

Calculate the stress from the combined effects of thermal gradients, pressure, and mechanical loads. Provide sketches or free body diagrams that show the configuration and dimensions of the members or bodies being analyzed, and locate the points at which the stresses are being calculated. Consider in the analysis whether repeated cycles of thermal loadings, together with other loadings, will cause fatigue failure or extensive accumulations of deformation. Make the appropriate stress combinations and compare the resulting stresses with design criteria (NRC 1989a). Consider the effects of underground temperature variations on the engineered barrier system (excluding the waste package) and its components.

As required by 10 CFR 60.21(c)(1)(ii)(D), provide a comparative evaluation of alternative engineered barrier system (excluding the waste package) designs and design concepts, with regard to design features that would be beneficial from a structural standpoint. There is no need to show that the design chosen is the best possible from the standpoint of isolation or from any other perspective. Rather, it is necessary to show a defensible rationale, including safety considerations, practicality, and cost considerations, explaining why the alternatives were not chosen. The author(s) must show that due diligence was exercised by DOE in investigating potential designs and determining which structural design features best serve the public and national interests. Structural considerations are a subset of all conditions that factor into selection of the engineered barrier system (excluding the waste package) design. The discussion of alternatives in this subsection should refer to the discussion that describes the overall assessment of alternative designs for the engineered barrier system (excluding the waste package) based on consideration of all appropriate factors.

6.2.3.5 Engineered Barrier System (Excluding the Waste Package) Shielding

Describe in this section the evaluation of the compliance of the engineered barrier system (excluding the waste package) and its components with the shielding design criteria. Summarize the adequacy of the shielding design features of the engineered barrier system (excluding the waste package) (if applicable) in accordance with 10 CFR 60.111(a) and 10 CFR 60.131(a)(3). The performance objective of 10 CFR 60.111(a) is mostly addressed in Chapter 7, but aspects applicable to impact of the engineered barrier system (excluding the waste package) on performance should be discussed in this section of Chapter 6. Provide a summary table showing gamma and neutron dose rates with the engineered barrier system (excluding the waste package) in place, under normal and accident conditions. Describe the interface between the design evaluation described in this section and the overall repository preclosure and postclosure performance assessments described in Chapters 7 and 8.

Show that the engineered barrier system (excluding the waste package) design will allow its radiation shielding features to meet applicable dose requirements for workers. Verify that the source terms used for engineered barrier system (excluding the waste package) radiation calculations are consistent with those provided in Chapters 5 and/or 10 of the Safety Analysis Report. Show that temperature-sensitive shielding materials will not be subjected to temperatures at or above their design limitations during normal, off-normal, or accident conditions.

Provide a general description of the basic method used to determine the gamma and neutron dose rates at the selected points outside the engineered barrier system (excluding the waste package).

Include a description of the spatial source distribution and any computer program used, with its documentation referenced. Discuss basic input parameters in detail. 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. For each program, provide the following information to assist the NRC reviewer in determining its applicability and validity (NRC 1997a, Section 5.V.4.a):

- Author, source, dated version, and facility.
- Description of the program, and the extent and limitation of its application.
- Reference to a series of test problems, demonstrating substantial similarity to solutions obtained from any one of the following: hand calculations, analytical results published in literature, experimental tests, a similar program, or benchmark problems. Prepare a summary comparison of the solutions obtained using the selected source, in either graphical or numeric form, and make it available in a reference. Justify any significant deviations between the model and the results from other sources.

Show that the number of dimensions of the code is appropriate for the dose rates being calculated. Generally, at least a two-dimensional calculation is necessary because one-dimensional codes provide little information about off-axis locations and streaming paths that may be significant for determining occupational exposure (NRC 1997a, Section 5.V.4.a).

If the author desires and this information is provided in other chapters of the License Application, the discussion of these topics in Chapter 6 may consist mainly of a reference to the locations in the License Application at which the information may be found (NRC 1989a).

Verify that shielding model dimensions and materials are consistent with those specified in the Safety Analysis Report drawings. Show that voids, streaming paths, and irregular geometries are accounted for or otherwise treated in a conservative manner. Show or describe locations selected for various dose calculations, and show that these dose points are representative of all locations relevant to shielding issues.

As required by 10 CFR 60.21(c)(1)(ii)(F), describe the modeling methods used to demonstrate that the design parameters are met. Provide an explanation of the measures supporting the models used to perform the analyses. Provide support for analyses and models used to predict future conditions and changes in the engineered barrier system (excluding the waste package) environment. Include in the Safety Analysis Report a representative computer code input file. Verify that information from the shielding model is properly entered into the code. Also, show that the cross-section library used by the code is appropriate for the selected application (NRC 1997a, Section 5.V.4.a).

In NUREG-1536 (NRC 1997a) NRC states that it accepts use of ANSI/ANS 6.1.1-1977 (ANSI/ANS 1977) for flux-to-dose rate conversions but states that a later version of the same standard (ANSI/ANS 6.1.1-1991) (ANSI/ANS 1991) may provide significantly lower calculated dose rates than those calculated from the earlier version of the code or those determined on the

basis of 10 CFR 20. State the version of the code used; if the 1991 version is used, justify its use given the apparent concern by NRC about neutron dose rates calculated using this version (NRC 1997a, Section 5.V.4.a).

Provide an explanation of measures supporting the models. For both design applications and accident analyses, provide an explanation of measures supporting the models used to perform analyses. Support analyses and models that have been used to predict future conditions and changes in the system 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.

Discuss the variability and uncertainty of data and the propagation of errors. Include evaluations of data representativeness, as well as uncertainties associated with the extrapolation of data. Discuss conceptualizations and the documentation and validation of codes and models used with respect to uncertainties related to the data on which the model is based, the applicability of specific models, the appropriateness of assumptions used in modeling, and the sensitivity of model results to the uncertainty of the input data. Input and output data and interpretations also should be provided along with the basis for the interpretation. Document the role of expert judgment if used.

Based on anticipated preclosure conditions, provide estimates of collective doses per year and dose rates associated with emplacement operations, including handling, maintenance, inspection, repair, and decommissioning, or refer the reader to Chapters 7 and 10 in the License Application that contain this information. Dose rates should be provided for all accessible locations during these operations.

Discuss neutron activation of engineered barrier system (excluding the waste package) components. Reference Chapter 5 for the gamma decay source strength as a function of photon energy and for the neutron source strength (NRC 1989a).

Reference the model used in the shielding calculation in Chapters 7 and 10. Include sketches to scale and dimensions of the radial and axial shielding materials. Show dose point locations for the various calculations external to the engineered barrier system (excluding the waste package). Discuss in detail voids or irregularities not taken into account in the model, showing that the resultant dose rates are conservative. Clearly identify differences in the models for normal and accident conditions. Provide material densities (g/cm) and the atomic number densities (atoms/barn-cm) for constituent nuclides of all materials used in the calculational models for the normal and accident analyses. Reference sources of data (NRC 1989a).

Discuss the analyses or tests to be performed to ensure adequate shielding for both gamma and neutron sources. Describe the acceptance criteria as well as the action to be taken if the criteria are not met (NRC 1989a).

As required by 10 CFR 60.21(c)(1)(ii)(D), provide a comparative evaluation of alternative engineered barrier system (excluding the waste package) designs and design concepts, with regard to design features that would be beneficial from a shielding standpoint. There is no need to show that the design chosen is the best possible from the standpoint of isolation or from any

other perspective. Rather, it is necessary to show a defensible rationale, including safety considerations, practicality, and cost considerations, explaining why the alternatives were not chosen. Author(s) must show that due diligence was exercised by DOE in investigating potential designs and determining which shielding design features best serve the public and national interests. Shielding considerations are a subset of all conditions that factor into selection of the engineered barrier system (excluding the waste package) design. The discussion of alternatives in this subsection should refer to the discussion that describes the overall assessment of alternative designs for the engineered barrier system (excluding the waste package) based on consideration of all appropriate factors.

V. REFERENCES

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

10 CFR 60. Energy: Disposal of High-Level Radioactive Wastes in Geologic Repositories.

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APPENDIX A

CROSS-REFERENCE BETWEEN 10 CFR 60 REGULATIONS AND TECHNICAL GUIDANCE DOCUMENT SECTIONS

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APPENDIX A

CROSS-REFERENCE BETWEEN 10 CFR 60 REGULATIONS AND TECHNICAL GUIDANCE DOCUMENT SECTIONS

This Appendix provides a cross-reference between 10 CFR 60 regulations and the sections in the *Technical Guidance Document for License Application Preparation* that provide guidance for demonstrating compliance with those regulations. All section references are the License Application section(s) in which compliance is to be demonstrated and the corresponding technical guidance document subsection (under Section IV) that provides acceptance criteria and guidance for the License Application section(s).

Not all regulations in 10 CFR 60 actually contain requirements for the licensee. Therefore, notes are provided to identify the reasons why these regulations are not addressed in the document. The following notes are referenced in the cross-reference that follows:

Note 1: Not a requirement; not addressed in the technical guidance document.

Note 2: Definition of anticipated processes and events contains a required assumption that is discussed in Chapter 8 of this document.

Note 3: Not a requirement needing to be specifically addressed in the License Application, or a general requirement not applicable to one or more specific sections in the License Application; not addressed in the technical guidance document.

Note 4: Not a requirement of the licensee; may be referred to but is not required to be addressed in the technical guidance document.

Note 5: To be addressed in the License Application sections of Chapter 4 as indicated; however, discussion of the regulation in the technical guidance document appears in the General Guidance section of Chapter 4 or in Section 4.0 to prevent repetition of guidance.

Note 6: Requirement not applicable to license to construct; not addressed in this revision of the technical guidance document.

Note 7: Not applicable to an unsaturated site like Yucca Mountain; may be referred to but is not required to be addressed in the technical guidance document.

10 CFR 60 Item	Technical Guidance Document/License Application Chapter and Section
60.1	Note 1
60.2	Note 1; Note 2; Anticipated and Unanticipated Processes and Events - 8.2.4, 13.3
60.3	Note 3
60.4(a)	Note 3
60.4(b)	11.4
60.5	Note 3
60.6	Note 1
60.7	Note 1
60.8	Note 1
60.9	11.2
60.10	Note 3
60.11	Note 3
60.15	Note 3
60.16	Note 3
60.17	Note 3
60.18	Note 3; Note 4
60.21(a)	1.1, 1.2, 1.3, 1.4, 1.5, 1.6, 1.7
60.21(b)(1)	1.1, 1.3, 4.0, 4.3
60.21(b)(2)	1.5
60.21(b)(3)	1.6
60.21(b)(4)	1.6
60.21(b)(5)	1.7
60.21(c)(1)	4.0, 4.1, 6.1
60.21(c)(1)(i)	4.3
60.21(c)(1)(i)(A)	3.3.6, 3.3.7, 3.3.9, 8.2.1
60.21(c)(1)(i)(B)	3.3.6, 3.3.7
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60.21(c)(1)(i)(F)	3.3.8, 3.3.9, 3.5.3, 3.6, 3.7, 3.7.1, 3.7.2, 3.7.3, 4.4
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60.21(c)(1)(ii)(B)	3.3.3, 3.3.4, 3.3.5, 3.3.7, 3.3.8, 3.5.1.5, 3.5.2.2, 3.5.3.5, 3.5.3.6, 3.9, 8.3
60.21(c)(1)(ii)(C)	3.8, 6.2.3, 8.3, 8.4, 8.4.1
60.21(c)(1)(ii)(D)	3.7, 3.8, 3.10, 4.0, 4.3, 4.5, 5.1.4, 5.2.3, 5.3.3, 5.4.3, 5.5.3, 5.6.3, 6.0, 6.2, 6.2.2, 6..2.3, 8.3.1
60.21(c)(1)(ii)(E)	4.1, 4.2, 4.3, 4.4, 4.5, 4.6 (Note 5), 5.1.4, 6.2.2, 6.2.3, 9.2.2, 9.3.2, 9.4.2, 9.5.2, 9.6.2, 12.5
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60.21(c)(2)(iv)	4.1, 4.2, 4.3, 4.4, 4.5, 4.6 (Note 5), 6.2.2
60.21(c)(3)	4.1, 4.3, 4.5, 4.6 (Note 5), 5.1, 6.2.2, 9.2.2, 9.3.2, 9.4.2, 9.5.2, 9.6.2, 12.5
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60.21(c)(12)	4.1, 4.3, 6.2.2, 11.1.6
60.21(c)(13)	3.3.10
60.21(c)(14)	4.0, 4.1, 4.2, 4.3, 4.4, 4.5 (Note 5), 5.1.4, 6.2.3, 11.10, 12.7
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60.21(c)(15)(iii)	11.2
60.21(c)(15)(iv)	4.1, 4.2, 4.3 (Note 5), 11.1
60.21(c)(15)(v)	4.3, 10.1.2, 10.4.2, 11.1
60.21(c)(15)(vi)	4.5, 4.6, 10.1.3
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60.111(b)(3)	2.1, 4.3.5
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60.113(a)(1)(i)(A)	2.1, 8.3.1
60.113(a)(1)(i)(B)	2.1, 8.3.1
60.113(a)(1)(ii)(A)	2.1, 8.3.1, 8.4.3
60.113(a)(1)(ii)(B)	2.1, 8.3.1, 8.4.3
60.113(a)(2)	2.1, 8.3.2
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60.121(a)	13.0, 13.1, 13.1.1, 13.1.3
60.121(a)(1)	2.1
60.121(a)(2)(i)	2.1
60.121(a)(2)(ii)	2.1
60.121(a)(2)(iii)	2.1
60.121(b)	2.1, 13.0, 13.2, 13.2.3
60.121(c)	2.1, 13.0, 13.1, 13.1.4, 13.2.3
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60.122(a)(2)(i)	2.1, 3, 3.10, 8.4
60.122(a)(2)(ii)	2.1, 3.10, 8.3.2
60.122(a)(2)(iii)(A)	2.1, 3.10, 8.3.2, 8.4
60.122(a)(2)(iii)(B)	2.1, 3.10, 8.3.2, 8.4
60.122(a)(2)(iii)(C)	2.1, 3.10
60.122(b)(1)	2.1, 3.3, 3.3.5, 3.3.7, 3.4, 3.5, 3.5.2, 3.5.3, 8.3.2
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60.122(b)(4)	2.1, 3.6, 3.6.1, 3.6.3, 3.7, 3.7.3, 8.3.2
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60.122(b)(6)	2.1, 3.1.2, 3.4, 3.5.1.7, 3.5.1.8, 3.5.2.9, 3.5.3.6, 8.3.2
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60.122(b)(8)(i)	2.1, 3.4, 3.4.1.1, 3.5.3, 3.5.3.4, 8.3.2
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60.122(c)(1)	2.1, 3.5, 3.5.1, 3.5.1.3, 3.5.1.5, 8.2.1, 8.3.2
60.122(c)(2)	2.1, 3.5, 3.5.2.8, 3.5.2.9, 3.5.3.6, 8.2.1, 8.3.2
60.122(c)(3)	2.1, 3.3.3, 3.5, 3.5.1, 3.5.2.2, 3.5.3.1, 8.2.1, 8.3.2
60.122(c)(4)	2.1, 3.3, 3.3.7, 3.3.9, 3.5, 3.5.2.2, 8.2.1, 8.3.2
60.122(c)(5)	2.1, 3.3.9, 3.4, 3.5, 3.5.1, 3.5.1.6, 3.5.2.2, 3.5.3.5, 3.7, 8.2.1, 8.3.2
60.122(c)(6)	2.1, 3.4, 3.4.2, 3.4.3, 3.5, 3.5.1.6, 3.5.2.2, 3.5.2.4, 3.5.3, 3.5.3.1, 8.2.1, 8.3.2
60.122(c)(7)	2.1, 3.6, 3.6.1, 3.6.2, 8.2.1, 8.3.2

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60.122(c)(9)	2.1, 3.6, 3.6.2, 8.2.1, 8.3.2
60.122(c)(10)	2.1, 3.3.6, 3.5.3, 3.5.3.3, 8.2.1, 8.3.2
60.122(c)(11)	2.1, 3.3, 3.3.4, 3.3.7, 3.5.2.2, 8.2.1, 8.3.2
60.122(c)(12)	2.1, 3.3, 3.3.4, 3.3.7, 8.2.1, 8.3.2
60.122(c)(13)	2.1, 3.3, 3.3.4, 3.3.7, 8.2.1, 8.3.2
60.122(c)(14)	2.1, 3.3, 3.3.4, 3.3.7, 8.2.1, 8.3.2
60.122(c)(15)	2.1, 3.3, 3.3.2, 3.3.3, 3.3.7, 8.2.1, 8.3.2
60.122(c)(16)	2.1, 3.3, 3.3.5, 3.5, 8.2.1, 8.3.2
60.122(c)(17)(i)	2.1, 3.3, 3.3.10, 8.2.1, 8.3.2
60.122(c)(17)(ii)	2.1, 3.3, 3.3.10, 8.2.1, 8.3.2
60.122(c)(18)	2.1, 3.3.10, 8.2.1, 8.3.2
60.122(c)(19)	2.1, 3.3.10, 8.2.1, 8.3.2
60.122(c)(20)	2.1, 3.3, 3.3.8, 3.5, 3.6, 8.2.1, 8.3.2
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60.122(c)(22)	2.1, 3.4, 3.5, 3.5.2.2, 3.5.2.6, 8.2.1, 8.3.2
60.122(c)(23)	2.1, 3.3.9, 3.4, 3.5, 3.5.2.6, 3.5.3, 3.5.3.4, 8.2.1, 8.3.2
60.122(c)(24)	2.1, 3.5, 3.5.3, 3.6, 3.6.3, 3.7, 8.2.1, 8.3.2
60.130	2.1, 4.1, 4.2, 4.3, 4.4, 4.5 (Note 5), 6.2.2, 9.2.1, 9.3.1, 9.4.1, 9.5.1, 9.6.1, 10.2
60.131(a)	2.1, 4.1, 4.2, 4.3, 4.4, 4.5, 4.6 (Note 5), 7.2.3, 7.7.2, 7.8, 12.5
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60.131(a)(2)	2.1, 6.2.2, 9.2.1, 9.3.1, 9.4.1, 9.5.1, 9.6.1, 10.2.1
60.131(a)(3)	2.1, 6.2.2, 6.2.3, 9.2.1, 9.3.1, 9.4.1, 9.5.1, 9.6.1, 10.2.2
60.131(a)(4)	2.1, 9.2.1, 9.3.1, 9.4.1, 9.5.1, 9.6.1, 10.2.3, 10.2.4
60.131(a)(5)	2.1, 10.2.1
60.131(a)(6)	2.1, 9.2.1, 9.3.1, 9.4.1, 9.5.1, 9.6.1, 10.2.4
60.131(b)	2.1, 4.3, 5.4, 6.2.2, 7.8, 9.2.1, 9.3.1, 9.4.1, 9.5.1, 9.6.1, 12.5
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60.131(d)	2.1, 4.1, 4.2, 4.3 (Note 5), 12.5
60.131(d)(1)	2.1, 4.1, 4.2, 4.3 (Note 5), 6.2.2, 7.8, 9.2.1, 9.3.1, 9.4.1, 9.5.1, 9.6.1
60.131(d)(2)	2.1, 4.1, 4.2, 4.3, 6.2.2, 9.2.1, 9.3.1, 9.4.1, 9.5.1, 9.6.1
60.131(d)(3)	2.1, 4.1, 4.2, 4.3, 7.8
60.131(d)(4)	2.1, 4.1, 7.8, 9.2.1, 9.3.1, 9.4.1, 9.5.1, 9.6.1
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60.131(e)(2)	2.1, 4.1, 4.3
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60.131(g)	2.1, 4.1, 4.2, 4.3, 6.2.2, 9.2.1, 9.3.1, 9.4.1, 9.5.1, 9.6.1, 12.5
60.131(h)	2.1, 4.1, 4.2, 4.3 (Note 5), 5.4, 6.2.3, 9.2.2

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60.132(a)	2.1, 4.1
60.132(b)	2.1, 4.1, 7.8, 10.2.3
60.132(c)(1)	2.1, 4.1, 7.8, 9.2.1, 9.3.1, 9.4.1, 9.5.1
60.132(c)(2)	2.1, 4.1, 9.6.1
60.132(d)	2.1, 4.1, 9.2.1, 9.3.1, 9.4.1, 9.5.1
60.132(e)	2.1, 4.1, 4.6, 9.2.1, 9.3.1, 9.4.1, 9.5.1, 9.6.1
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60.135(c)(3)	2.1, 5.1.3
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60.140(d)(4)	2.1, 12.1, 12.2, 12.3, 12.4, 12.6
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APPENDIX B

ACRONYMS

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APPENDIX B

ACRONYMS

ALARA	As Low As Reasonably Achievable
BLM	U.S. Bureau of Land Management
CFR	U.S. Code of Federal Regulations
CRWMS	Civilian Radioactive Waste Management System
DOE	U.S. Department of Energy
EPA	U.S. Environmental Protection Agency
HLW	High-Level Radioactive Waste
IRSR	Issue Resolution Status Report
KTI	Key Technical Issue
M&O	Management and Operating Contractor
MGR	Monitored Geologic Repository
MSHA	Mine Safety and Health Act
MTU	Metric Tons of Uranium
NEC	National Electrical Code
NEPA	National Environmental Policy Act
NMC&A	Nuclear Material Control and Accounting
NRC	U.S. Nuclear Regulatory Commission
NWPA	Nuclear Waste Policy Act
OCRWM	Office of Civilian Radioactive Waste Management
OQA	Office of Quality Assurance
OSHA	Occupational Safety and Health Act
PISA	Project Integrated Safety Assessment
QA	Quality Assurance
QARD	*Quality Assurance Requirements and Description*
YMP	Yucca Mountain Site Characterization Project

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APPENDIX C

LICENSE APPLICATION WRITER'S GUIDE

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APPENDIX C

LICENSE APPLICATION WRITER'S GUIDE

C.1 INTRODUCTION

The purpose of this appendix is to guide License Application authors in document structure, format, and style; to provide for a consistent document; and to strengthen the production process. This writer's guide and the documents that it references will serve as the only references for the License Application structure, format, and style.

The basic organization and guidance on content of the License Application is provided in the body of this technical guidance document.

This writer's guide assumes that the License Application will be developed for hard copy distribution. Additional instructions are provided to address the possibility that the document may be published electronically (see Section C.6 of this appendix).

C.2 WRITE WITH THE AUDIENCE IN MIND

The License Application primarily seeks to clearly present the U.S. Department of Energy safety assessment for a potential high-level radioactive waste repository at Yucca Mountain. While the License Application will be submitted to the U.S. Nuclear Regulatory Commission (NRC), keep in mind that readers may include those in the political arena as well as the general public. Since the License Application will report complex technical subjects and phenomena, the challenge is to present these ideas in terms that any interested reader can understand. Minimize the use of jargon and complex technical expressions and provide explanations where complex expressions must be used. In addition, follow the guidance below:

- Use active rather than passive voice wherever possible to produce a stronger and more assertive document.
- Write in the third person rather than the first person.
- Use short declarative sentences; break up large sections.
- Avoid superlatives and exaggeration. A dry, slightly understated position is more defensible.
- Be certain of the facts.

C.3 LICENSE APPLICATION DOCUMENT STRUCTURE

The basic organization of presentation of material in the License Application is provided in the body of the *Technical Guidance Document for License Application Preparation*. The document structure is hierarchical, starting with a general subject and leading to more specific subjects at the lower levels that support the higher-level topic. Present material in a logical, linear progression. Introductory statements at the beginning of each section will assist in establishing this structure.

C.3.1 Chapters

Each chapter will contain a table of contents consistent with the table of contents for the License Application that is provided in this technical guidance document. The table of contents will also show additional sections created by chapter authors at greater levels of indention than levels provided in this technical guidance document. Each chapter will contain lists of figures and tables, as appropriate.

C.3.2 Sections and Subsections

Within each chapter, use sections and subsections to organize text. Organizing the document into sections and subsections implements a philosophy of dividing topics into units that can be broken out individually for review. There must always be at least two subsections within a section. In other words, there cannot be a 2.1.1 without a 2.1.2.

C.3.3 Sections

A section is a unit of text residing at the second level of indention. Number 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 this technical guidance document unless a different organization is approved, in writing, by the Management and Operating Contractor (M&O) Licensing Manager.

C.3.4 Subsections

Subsections reside at the third and lower levels of indention. Number subsections using three or four digits separated by periods (e.g., x.x.x, x.x.x.x) depending upon the level of indention.

Create subsections as follows:

2.1.1

2.1.1.1

2.1.1.2

2.1.2

The subsection structure for the License Application will be consistent with the License Application structure provided in this technical guidance document, although authors may create more subsections. The NRC *License Application Review Plan for a Geologic Repository for Spent Nuclear Fuel and High-level Radioactive Waste*, NUREG-1323, and/or the *Mined Geologic Disposal System License Application Annotated Outline,* YMP/94-05 (DOE Office of Civilian Radioactive Waste Management, in draft), are available from the M&O Licensing Manager for examples of subsection structure for respective chapters.[1] These documents provide a subsection structure that corresponds to NRC guidance for a repository License Application. Use of the *Mined Geologic Disposal System License Application Annotated Outline* structure is suggested, not required.

Subsection numbering is limited to the fourth level of indention (e.g., 2.1.1.1). If necessary, further divide text using zero level indention, as described in Subsection C.3.5.

In past documents of this type, flexibility in subsection indention has led to substantial inconsistency between subsections completed by different authors. Although there are no hard rules for when to create new subsections, use the following general guidance:

- Although use of numbered subsections below the fourth level of indention (x.x.x.x) is not allowed, unnumbered zero level indention (as described in Subsection C.3.5) is permitted when discussing complex topics, such as site characteristics.

- Do not create a new subsection if the section consists of one or two paragraphs. If it is necessary to further detail the topic, use zero level of indention titles.

- Ensure that the lower level subsection logically expands upon the higher level subsection.

- Use zero level of indention titles (see Subsection C.3.5) to relate text to items that are best described in a list.

C.3.5 Zero Level of Indention

A zero level of indention heading contains no section number. The heading text is terminated with a period and two spaces. The heading text, which is bolded and left justified, is embedded within the first paragraph of the zero-level subsection. The first letter of each word in the subsection title is upper case. Because zero-level subsections are not numbered, they do not appear in the table of contents. The following illustrates a zero level of indention heading:

[1] Although NUREG-1323 is dated, it provides the latest guidance from NRC regarding format for the License Application.

Example:
Characteristics of Earthquake Ground Motions at Yucca Mountain. To date, earthquake ground motions at Yucca Mountain have been estimated using attenuation...

C.3.6 Lists

Use bullets for lists. Lists can include complete sentences; however, if each list item becomes a lengthy paragraph, the preferred style might be a series of subsections rather than a list. Use a list to mention a series of items that are an integral part of a discussion.

C.3.7 Definitions

A glossary will be included for each chapter. While developing your text, identify those terms that might not be easily understood by the audience. Write the preferred definition for the term, and provide a list of terms and definitions to Technical Publications Management for inclusion in the chapter glossary.

C.3.8 References

Format references in accordance with the M&O *Publishing Style Guide*. Include the applicable page number(s) from the original source. In addition, all the License Application references are required to have either a Records Information System accession number, a Technical Information Center catalog number, quality assurance status, or a data tracking number (if applicable) at the end of the complete reference description in the reference section of the chapter in which the reference is cited. Data tracking numbers are required for references to data in the GENISES or Reference Information Base databases.

Authors are strongly encouraged to assemble a reference package that contains copies of the first two or three pages of the document referenced. These pages should show the title, author, and publishing information. The reference packages will be used by researchers finding documents in the Records Information System or Technical Information Center, and by authors and editors who will be verifying titles and other reference information.

C.3.9 Cross-Referencing

Cross-referencing is encouraged to reduce the amount of duplicate information and to minimize the chance of presenting contradictory information. To cross-reference another author's material:

- Inform the other lead author of the cross-reference.

- Verify during final preparation of the chapter for submittal to the M&O Licensing Manager that the cross-reference to the other author's work is still valid and correctly numbered.

- Use the term "Section" rather than "Subsection" for all cross-referencing (e.g., Section 3.2.2.5).

C.4 LICENSE APPLICATION DOCUMENT TEXT FORMAT

Use Microsoft Word to prepare your text. Use the default settings for margins, justification, spacing, font, and tab settings.

C.4.1 Style

Technical Publications Management and M&O Licensing are responsible for the overall style and appearance of the License Application. Direct any questions regarding style or format to Technical Publications Management or M&O Licensing.

C.4.2 Figure and Table Numbering

Number figures and tables according to the second level heading. In other words, the first figure referred to in Chapter 2, Section 2.3, is Figure 2.3-1. Number figures and tables separately. Place figures and tables at the end of each chapter; first tables, then figures.

C.4.3 Formatting Figures and Maps

Ensure that all text provided in figures is legible. See the M&O *Publishing Style Guide* for further guidance on how to format figures. All maps proposed for inclusion in the License Application must be processed through M&O Technical Data Management. The preferred projection is Universal Transverse Mercator. However, if necessary, obtain permission from M&O Technical Data Management to use a different projection.

C.4.4 Formatting Tables

Ensure that all text provided in tables is legible. See the M&O *Publishing Style Guide* for further guidance on how to format tables. Clearly identify tables that contain quality data.

C.4.5 Headers and Footers

Technical Publications Management will insert footers. The footer will include the License Application document number in the lower left corner. The date of the page, which can vary from page to page, will be contained in the lower right corner. If the page is in draft form, the footer will so indicate by "DRAFT, xx/xx/xx" where xx/xx/xx is the date of the draft. Both the document number and the date will be in 10-point type. Although M&O Licensing may elect to insert a header, one is presently not planned.

C.4.6 Units of Measurement

Although the units of the International System of Units are becoming more common in the United States, some readers do not understand them. Therefore, measurements expressed in the International System of Units will be expressed in both SI and English units, first by International System of Units and immediately followed by the English equivalent in parentheses, 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.
- For measurements commonly expressed in English units, such as the diameter of pipes, English units are used without conversion to the International System of Units.
- Quantities on maps, such as elevations, given in English units are not converted to the International System of Units quantities.

Certain quantities may customarily be expressed in mixed units, such as English and the International System of Units, as in the case of metric tons heavy metal per acre. This practice is undesirable and should be avoided; nevertheless, the author may choose to use mixed units if use is predominant and if the use of other units would not add clarity or assist in understanding the meaning of the quantity.

C.4.7 Numbers

All numbers that appear before units of measurement are written as figures.

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; write "0.5 meter."

In expressing a range or series of measurements, do not repeat the units; write "40 to 50°C" and "5 and 10 rem," or "40, 60, or 90 cm."

Numbers in text are spelled out if they are fewer than 10 or if they begin a sentence. If any number in a series is greater than 10, the entire series is written as figures.

Fractions standing alone are spelled out, "two-thirds of the site." Fractions that are not spelled out are best expressed as decimals rather than fractions (e.g., 3.75 rather than 3 3/4).

Avoid changing units unnecessarily when reporting different amounts of the same quantity, for example, changing units of radiation dose from rem to millirem in a discussion.

C.4.8 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 be used for angles, compass directions, longitude, and latitude. In text, spell out percent. Use the percent symbol (%) in tables or figures.

Use standard abbreviations for units of measure. Do not follow abbreviations with a period. If the abbreviation is derived from the name of a person (e.g., W.K.), it is uppercase; otherwise it is lowercase (i.e., m, 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 (G and M, respectively).

When the measure is a compound unit designating the multiplication of one unit by another, indicate multiplication by a hyphen (e.g., g-cm, W-s); indicate division by the slash symbol (e.g., J/mole-K, kcal/m-s-K). Write measurements involving powers in the exponential form (e.g., 10 m^3, 8.34 x 10^{-8}).

In reference to radioisotopes in text, write Cesium-137 instead of ^{137}Cs. In tables, write Cs-137. In tables, use the superscript form only when there is no room for the longer form.

C.5 STYLE

Refer to the M&O *Publishing Style Guide* for style information applicable to the License Application that is not specifically discussed in this writer's guide. Pay particular attention to Chapters 9, 10, and 11 of the M&O *Publishing Style Guide*.

In addition to the guidance provided in the M&O *Publishing Style Guide*, the following guidance applies.

Syntax. Writers must be particularly alert to syntax and choice of verbs to avoid inadvertently undermining the completed work. There is a spectrum of certainty implicit in writers' syntax. Writers should use a word that fits the intended meaning, but should seek to make syntax choices using high confidence words when possible. Low confidence words include the following: may, maybe, might, could be, seem, appear, suggest, imply, infer, deduce, expect, assume, conceivable, probably, likely, possibly. Examples of high confidence words are: illustrates, concludes, shows, resolves, states, demonstrates, indicates, establishes, documents, proves.

"Relatively" and "significant" are words that confuse and must be used sparingly, if at all. "The impacts are relatively harmless." The reader must ask, "Relative to what?" "The U-series dating technique is significantly better than the U-trend technique." The reader must ask, "Significant according to what standard?"

Commonly Misused Phrases and Words. The M&O *Publishing Style Guide* contains guidance on troublesome words. Additional 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 forms. The corresponding singular forms are "datum," "medium," "phenomenon," and "criterion."

- The words "offsite" and "onsite," written as single words, are used as adjectives, not as adverbs. "The plans call for onsite processing" is acceptable. "Processing is performed onsite" is not acceptable; a phrase like "at the site" must replace "onsite."

- The adverbial phrase "under way," 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 the most careful writers; it is not a substitute for "because of." Use it only when "due" 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 on 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 guess 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."

- Do not use the slash symbol (/) to mean "and." The slash should be used only to denote division in units of measurement. Do not use "and/or."

Vogue and Vague Words. Some words and phrases are in such common use among writers of program documents that they are often used imprecisely or even with no meaning at all.

- "Anticipate." This word is not a synonym for "expect."

- "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.

- 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. Some of these words and phrases have precise meanings, but they are pretentious. Do not use them.

- "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, it requires explanation by pointing out explicitly which parts of the analysis produce the overestimates. Giving such a complete definition of the word, however, usually removes the need for it.

- "Consider" and "factor." These words are vague, although "factor" does have a precise meaning in mathematics. Writers use them to mean "criterion," "design specification," or something to think about.

- "Facility." This word usually conveys little information; define it more clearly.

- "Ologies." The indiscriminate use and coining of words ending in "ology" leads to imprecise writing. In careful use, the suffix is reserved for words that express 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 the meaning. Do not use "methodology" to mean "methods."

- "Orders of magnitude." This phrase is almost incomprehensible to people who do not use technical jargon frequently. Write "one ten-thousandth of x" or "10,000 times smaller than x" instead of "four orders of magnitude smaller than x."

C.6 ELECTRONIC DOCUMENT PUBLISHING

Should the License Application be published electronically rather than on paper, the following guidance applies:

- Hypertext links: To improve access times for the reader using Web browsers, figures and tables will be hypertext linked instead of appearing in the section. Items in the reference list will also be linked to images or header information in the Records Information System, Technical Information Center, or Technical Data Management System. For this reason, each cited reference must have an accession number, catalog number, or data tracking number.

- Figures and maps: Guidance for preferred electronic file formats will be provided separately. Figures and maps, especially those originating as computer-aided draft design drawings or computer models, present unique challenges when publishing electronically. Identify the number, types, sources, and file formats of figures and maps and provide this information to Web Publishing Management as early as possible during text development.

- Tables: Tables more than two pages long should be developed and maintained as a separate document file.

- Headers and footers: Electronic format may not allow headers and footers. Headers and footers for electronic publication will be developed separately by Web Publishing Management in cooperation with Technical Publications Management.

- Page numbering: In an electronic environment, documents do not have page numbers or "intentionally left blank" pages.

- Equations and Greek letters: Because Lotus Notes (used for electronic publishing) uses the same equation editor as Microsoft Word, create equations with Word's equation editor.

C.7 COPYRIGHTED INFORMATION

Because of the legal implications involving the quoting of information from copyright-protected sources, such quotations shall not be used in the License Application. In general, copyright-protected sources may be referenced only. Complete reproductions of text, tables, or figures from copyrighted sources shall not be used in the License Application until the necessary permissions are obtained by the Technical Information Center. Contact the Technical Information Center Supervisor for direction on obtaining copyright clearances.

CHAPTER 7. PRECLOSURE RADIOLOGICAL SAFETY ASSESSMENT

I. OBJECTIVE

This chapter provides guidance for the author(s) of Chapter 7 of the License Application. The License Application will be submitted to the U.S. Nuclear Regulatory Commission (NRC) for authorization to construct a geologic high-level radioactive waste repository at Yucca Mountain. This chapter discusses the information related to preclosure radiological safety analysis needed to support the License Application.

The application must demonstrate that the U.S. Department of Energy (DOE) has satisfactorily identified design basis events for the repository, categorized those events based on probability of occurrence, determined the predicted radiological consequences, and demonstrated that such radiological consequences are within prescribed regulatory requirements, both for repository workers and members of the general public.

II. GUIDANCE TOPICS

The following topics will be discussed in Chapter 7 of the License Application:

- 7.0 PURPOSE, SUMMARY, AND CONCLUSIONS
- 7.1 REQUIREMENTS AND SCOPE
- 7.2 APPROACH
  - 7.2.1 Event Type
  - 7.2.2 Radionuclide Releases
  - 7.2.3 Radionuclide Transport and Radiation Dose
- 7.3 REPOSITORY DESIGN INPUT TO SAFETY ASSESSMENT
  - 7.3.1 Repository Description
  - 7.3.2 Waste Package Description
- 7.4 SOURCE TERM DEVELOPMENT
  - 7.4.1 Light Water Reactor Spent Nuclear Fuel
  - 7.4.2 High-Level Radioactive Waste
  - 7.4.3 Radioactive Effluents and Wastes
  - 7.4.4 Other U.S. Department of Energy Spent Nuclear Fuel (if needed)
  - 7.4.5 Plutonium Disposition (if needed)
- 7.5 DESIGN BASIS EVENT DEFINITION
  - 7.5.1 Design Basis Event Categorization
  - 7.5.2 Category 1 Design Basis Events
  - 7.5.3 Category 2 Design Basis Events
- 7.6 POTENTIAL RELEASES FROM DESIGN BASIS EVENTS
  - 7.6.1 Initiating Events and Accident Sequence
  - 7.6.2 Repository Radionuclide Transport Mechanisms
- 7.7 ATMOSPHERIC DISPERSION AND DOSE CALCULATIONS
  - 7.7.1 Meteorological Data and Radionuclide Dispersion
  - 7.7.2 Radiological Dose Estimates

7.8 CONCLUSIONS
7.8.1 Category 1 Design Basis Events - Demonstration of Compliance to 10 CFR 60.111(a)
7.8.2 Category 2 Design Basis Events - Demonstration of Compliance to 10 CFR 60.136(b)

III. REGULATORY REQUIREMENTS

***** IMPORTANT NOTE *****

At the time of development of this document, it appears likely that NRC will issue a new, site-specific regulation for Yucca Mountain in 1999. At the time the new 10 CFR 63 is issued, 10 CFR 60 will be modified to state that it is not applicable to Yucca Mountain. Details about the new regulation are not known. However, it has been determined prudent to develop the working draft License Application in accordance with the best available estimate of the form of the new regulation. To avoid confusing references and to allow production of the technical guidance document in time to support training for the working draft License Application, the text of the technical guidance document that follows this note has not been revised to reflect the postulated form of the new regulation. Instead, this note provides guidance needed by the working draft License Application authors to comply with management direction on addressing the new regulation. The paragraphs that follow explain changes in the regulation with respect to the existing 10 CFR 60 that affect this technical guidance document chapter and provide guidance on how to address the changes in the working draft License Application. Where guidance in this note conflicts with guidance in the remainder of this chapter of the technical guidance document, the guidance in the note is to govern. Any questions should be addressed to M&O Licensing.

Changes to Regulations Applicable to Chapter 7

The following 10 CFR 60 requirements applicable to technical guidance document Chapter 7 and listed below in Section III are assumed to not be carried over to the new regulation: 60.130, 60.131, 60.132, and 60.133.

In place of these specific requirements, the new regulation is expected to require a discussion of how overall performance objectives for the repository are achieved in preclosure operations of the surface and subsurface facilities. The manner in which that contribution is to be described has not been specified, though an NRC review plan on the subject may be issued sometime in the future.

Guidance to the Working Draft License Application Authors for Addressing Changes

- Disregard guidance in Section IV regarding demonstrating compliance with any of the regulations assumed to be deleted as described above. Instead, discuss how the

performance and operational features of the waste package and surface and subsurface facilities support overall repository performance objectives for preclosure.

- Address the technical issue of each of the regulations in Section III, including those expected to be deleted, but do not relate the discussion to demonstrating compliance with any of the deleted regulations. Continue to explicitly address the regulations in Section III that are not expected to be deleted.

- Detailed discussion of design criteria, design bases, and performance is still required as per the existing guidance in Section IV. Discussion should still include detailed discussions of the design criteria and/or operational features of the waste package and surface and subsurface facilities that affect that performance.

- No sections or subsections of Chapter 7 are expected to be deleted as a result of this note.

***** END OF NOTE *****

Sections of the regulations that are to be addressed fully or in part in Chapter 7 of the License Application are reproduced below. These are verbatim extracts from the regulations. Paragraph numbers refer to Title 10 of the U.S. Code of Federal Regulations (CFR). In addition, Appendix A provides a cross-reference between the 10 CFR 60 regulations and the specific License Application sections that address these regulations. Revisions to regulations are not uncommon. Verify that current versions are used.

Both 10 CFR 20 and 10 CFR 60 have "Definitions" sections (10 CFR 20.1003 and 10 CFR 60.2, respectively) that precisely define numerous terms. The listing in 10 CFR 20.1003 is extensive, and the level of exactitude is noteworthy; for example, the word "week" is defined to mean exactly "seven consecutive days starting on Sunday." All of the various dose terms, such as absorbed dose, dose equivalent, effective dose equivalent, committed dose equivalent, committed effective dose equivalent, and total effective dose equivalent, are also defined in 10 CFR 20.1003. 10 CFR 60.2 provides specific definitions for the terms "design basis events" and "important to safety." It is critical that the License Application author review these listings of definitions, in their entirety, and ensure that the use of terms in the License Application be completely consistent therewith.

10 CFR 20, Standards For Protection Against Radiation
Subpart B - Radiation Protection Programs
20.1101 Radiation protection programs.

(d) To implement the ALARA requirements of § 20.1101 (b)[1], and notwithstanding the requirements in § 20.1301 of this part, a constraint on air emissions of radioactive material to the

[1] 10 CFR 20.1101(b) states: "The licensee shall use, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA)."

environment, excluding Radon-222 and its daughters, shall be established by licensees other than those subject to § 50.34a[2], such that the individual member of the public likely to receive the highest dose will not be expected to receive a total effective dose equivalent in excess of 10 mrem (0.1 mSv) per year from these emissions...

Subpart C - Occupational Dose Limits
20.1201 Occupational dose limits for adults.

(a) The licensee shall control the occupational dose to individual adults, except for planned special exposures under § 20.1206, to the following dose limits.

(1) An annual limit, which is the more limiting of-

(i) The total effective dose equivalent being equal to 5 rems (0.05 Sv); or

(ii) The sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye being equal to 50 rems (0.5 Sv).

(2) The annual limits to the lens of the eye, to the skin, and to the extremities, which are:

(i) An eye dose equivalent of 15 rems (0.15 Sv), and

(ii) A shallow-dose equivalent of 50 rems (0.50 Sv) to the skin or to any extremity.

(c) The assigned deep-dose equivalent and shallow-dose equivalent must be for the part of the body receiving the highest exposure...

(d) Derived air concentration (DAC) and annual limit on intake (ALI) values are presented in table 1 of Appendix B to Part 20 and may be used to determine the individual's dose (see § 20.2106) and to demonstrate compliance with the occupational dose limits.

(e) In addition to the annual dose limits, the licensee shall limit the soluble uranium intake by an individual to 10 milligrams in a week in consideration of chemical toxicity (see footnote 3 of Appendix B to Part 20).

(f) The licensee shall reduce the dose that an individual may be allowed to receive in the current year by the amount of occupational dose received while employed by any other person (see § 20.2104(e)).

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[2]10 CFR 50.34a states: "Design objectives for equipment to control releases of radioactive material in effluents-nuclear power reactors."

20.1202 Compliance with requirements for summation of external and internal doses.

(a) If the licensee is required to monitor under both § 20.1502(a) and (b), the licensee shall demonstrate compliance with the dose limits by summing external and internal doses. If the licensee is required to monitor only under § 20.1502(a) or only under § 20.1502(b), then summation is not required to demonstrate compliance with the dose limits. The licensee may demonstrate compliance with the requirements for summation of external and internal doses by meeting one of the conditions specified in paragraph (b) of this section and the conditions in paragraphs (c) and (d) of this section.

(Note: The dose equivalents for the lens of the eye, the skin, and the extremities are not included in the summation, but are subject to separate limits.)

(b) *Intake by inhalation.* If the only intake of radionuclides is by inhalation, the total effective dose equivalent limit is not exceeded if the sum of the deep-dose equivalent divided by the total effective dose equivalent limit, and one of the following, does not exceed unity:

(1) The sum of the fractions of the inhalation ALI for each radionuclide, or

(2) The total number of derived air concentration-hours (DAC-hours) for all radionuclides divided by 2,000, or

(3) The sum of the calculated committed effective dose equivalents to all significantly irradiated[3] organs or tissues (T) calculated from bioassay data using appropriate biological models and expressed as a fraction of the annual limit...

(d) *Intake through wounds or absorption through skin.* The licensee shall evaluate and, to the extent practical, account for intakes through wounds or skin absorption.

Note: The intake through intact skin has been included in the calculation of DAC for hydrogen-3 and does not need to be further evaluated.

20.1203 Determination of external dose from airborne radioactive material.

Licensees shall, when determining the dose from airborne radioactive material, include the contribution to the deep-dose equivalent, eye dose equivalent, and shallow-dose equivalent from

[3]An organ or tissue is deemed to be significantly irradiated if, for that organ or tissue, the product of the weighting factor, w_T, and the committed dose equivalent, $H_{T,50}$, per unit intake is greater than 10 percent of the maximum weighted value of $H_{T,50}$ (that is, $W_T H_{T,50}$) per unit intake for any organ or tissue.

external exposure to the radioactive cloud (see Appendix B to Part 20 footnotes 1 and 2).

Note: Airborne radioactivity measurements and DAC values should not be used as the primary means to assess the deep-dose equivalent when the airborne radioactive material includes radionuclides other than noble gases or if the cloud of airborne radioactive material is not relatively uniform. The determination of the deep-dose equivalent to an individual should be based upon measurements using instruments or individual monitoring devices.

20.1204 Determination of internal exposure.

(b) Unless respiratory protective equipment is used, as provided in § 20.1703, or the assessment of intake is based on bioassays, the licensee shall assume that an individual inhales radioactive material at the airborne concentration in which the individual is present.

(c) When specific information on the physical and biochemical properties of the radionuclides taken into the body or the behavior or the material in an individual is known, the licensee may-

(1) Use that information to calculate the committed effective dose equivalent...

(2) Upon prior approval of the Commission, adjust the DAC or ALI values to reflect the actual physical and chemical characteristics of airborne radioactive material (e.g., aerosol size distribution or density); and

(3) Separately assess the contribution of fractional intakes of Class D, W, or Y compounds of a given radionuclide (see appendix B to Part 201 [sic]) to the committed effective dose equivalent...

(e) If the identity and concentration of each radionuclide in a mixture are known, the fraction of the DAC applicable to the mixture for use in calculating DAC-hours must be either-

(1) The sum of the ratios of the concentration to the appropriate DAC value (e.g., D, W, Y) from Appendix B to Part 20 for each radio-nuclide in the mixture; or

(2) The ratio of the total concentration for all radionuclides in the mixture to the most restrictive DAC value for any radionuclide in the mixture.

(f) If the identity of each radionuclide in a mixture is known, but the concentration of one or more of the radionuclides in the mixture is not known, the DAC for the mixture must be the most restrictive DAC of any radionuclide in the mixture.

(g) When a mixture of radionuclides in air exists, licensees may disregard certain radionuclides in the mixture if-

(1) The licensee uses the total activity of the mixture in demonstrating compliance with the dose limits in § 20.1201 and in complying with the monitoring requirements in § 20.1502(b), and

(2) The concentration of any radionuclide disregarded is less than 10 percent of its DAC, and

(3) The sum of these percentages for all of the radionuclides disregarded in the mixture does not exceed 30 percent.

(h) (1) In order to calculate the committed effective dose equivalent, the licensee may assume that the inhalation of one ALI, or an exposure of 2,000 DAC-hours, results in a committed effective dose equivalent of 5 rems (0.05 Sv) for radionuclides that have their ALIs or DACs based on the committed effective dose equivalent.

(2) When the ALI (and the associated DAC) is determined by the nonstochastic organ dose limit of 50 rems (0.5 Sv), the intake of radionuclides that would result in a committed effective dose equivalent of 5 rems (0.05 Sv) (the stochastic ALI) is listed in parentheses in table 1 of Appendix B to Part 20. In this case, the licensee may, as a simplifying assumption, use the stochastic ALIs to determine committed effective dose equivalent. However, if the licensee uses the stochastic ALIs, the licensee must also demonstrate that the limit in § 20.1201(a)(1)(ii) is met...

20.1207 Occupational dose limits for minors.

The annual occupational dose limits for minors are 10 percent of the annual dose limits specified for adult workers in § 20.1201.

20.1208 Dose to an embryo/fetus.

(a) The licensee shall ensure that the dose to an embryo/fetus during the entire pregnancy, due to occupational exposure of a declared pregnant woman, does not exceed 0.5 rem (5 mSv)...

(b) The licensee shall make efforts to avoid substantial variation above a uniform monthly exposure rate to a declared pregnant woman so as to satisfy the limit in paragraph (a) of this section.

(c) The dose to an embryo/fetus shall be taken as the sum of-

(1) The deep-dose equivalent to the declared pregnant woman; and

(2) The dose to the embryo/fetus from radionuclides in the embryo/fetus and radionuclides in the declared pregnant woman.

(d) If the dose to the embryo/fetus is found to have exceeded 0.5 rem (5 mSv), or is within 0.05 rem (0.5 mSv) of this dose, by the time the woman declares the pregnancy to the licensee, the licensee shall be deemed to be in compliance with paragraph (a) of this section if the additional dose to the embryo/fetus does not exceed 0.05 rem (0.5 mSv) during the remainder of the pregnancy.

Subpart D - Radiation Dose Limits for Individual Members of the Public

20.1301 Dose limits for individual members of the public.

(a) Each licensee shall conduct operations so that-

(1) The total effective dose equivalent to individual members of the public from the licensed operation does not exceed 0.1 rem (1 millisievert) in a year, exclusive of the dose contributions from background radiation, from any medical administration the individual has received, from exposure to individuals administered radioactive material and released in accordance with § 35.75, from voluntary participation in medical research programs, and from the licensee's disposal of radioactive material into sanitary sewerage in accordance with § 20.2003, and

(2) The dose in any unrestricted area from external sources, exclusive of the dose contributions from patients administered radioactive material and released in accordance with § 35.75, does not exceed 0.002 rem (0.02 millisievert) in any one hour.

(b) If the licensee permits members of the public to have access to controlled areas, the limits for members of the public continue to apply to those individuals.

(c) A licensee or license applicant may apply for prior NRC authorization to operate up to an annual dose limit for an individual member of the public of 0.5 rem (5 mSv). The licensee or license applicant shall include the following information in this application:

(1) Demonstration of the need for and the expected duration of operations in excess of the limit in paragraph (a) of this section;

(2) The licensee's program to assess and control dose within the 0.5 rem (5 mSv) annual limit; and

(3) The procedures to be followed to maintain the dose as low as is reasonably achievable.

(d) In addition to the requirements of this part, a licensee subject to the provisions of EPA's generally applicable environmental radiation standards in 40 CFR part 190 shall comply with those standards...

20.1302 Compliance with dose limits for individual members of the public.

(b) A licensee shall show compliance with the annual dose limit in § 20.1301 by-

(1) Demonstrating by measurement or calculation that the total effective dose equivalent to the individual likely to receive the highest dose from the licensed operation does not exceed the annual dose limit; or

(2) Demonstrating that-

(i) The annual average concentrations of radioactive material released in gaseous and liquid effluents at the boundary of the unrestricted area do not exceed the values specified in table 2 of Appendix B to Part 20; and

(ii) If an individual were continuously present in an unrestricted area, the dose from external sources would not exceed 0.002 rem (0.02 mSv) in an hour and 0.05 rem (0.5 mSv) in a year.

(c) Upon approval from the Commission, the licensee may adjust the effluent concentration values in Appendix B to Part 20, table 2, for members of the public, to take into account the actual physical and chemical characteristics of the effluents (e.g., aerosol size distribution, solubility, density, radioactive decay equilibrium, chemical form).

10 CFR 60, Disposal of High-Level Radioactive Wastes in Geologic Repositories
Subpart A - General Provisions
60.2 Definitions.
As used in this part –
Design basis events means:

(1)(i) Those natural and human induced events that are reasonably likely to occur regularly, moderately frequently, or one or more times before permanent closure of the geologic repository operations area; and

(ii) Other natural and human induced events that are considered unlikely, but sufficiently credible to warrant consideration, taking into account the potential for significant radiological impacts on public health and safety.

(2) The events described in paragraph (1)(i) of this definition are referred to as "Category 1" design basis events. The events described in paragraph (1)(ii) of this definition are referred to as "Category 2" design basis events.

Important to safety, with reference to structures, systems, and components, means those engineered features of the repository whose function is:

(1) To provide reasonable assurance that high-level radioactive waste can be received handled, packaged, stored, emplaced, and retrieved without exceeding the requirements of § 60.111(a) for Category 1 design basis events; or

(2) To prevent or mitigate Category 2 design basis events that could result in doses equal or greater than the values specified in § 60.136 to any individual located on or beyond any point on the boundary of the preclosure controlled area.

Preclosure controlled area means that surface area surrounding the geologic repository operations area for which the licensee

exercises authority over its use, in accordance with the provisions of this part, until permanent closure has been completed.

Restricted area means an area, access to which is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials...

Unrestricted area means an area, access to which is neither limited nor controlled by the licensee.

Subpart B - Licenses
License Applications
60.21 Content of Application.

(a) An application shall consist of general information and a Safety Analysis Report...

(c)The Safety Analysis Report shall include:...

(3) A description and analysis of the design and performance requirements for structures, systems, and components of the geologic repository that are important to safety. The analysis must include a demonstration that–

(i) The requirements of § 60.111(a) will be met, assuming occurrence of Category 1 design basis events; and

(ii) The requirements of § 60.136 will be met, assuming occurrence of Category 2 design basis events-... (5) A description of the kind, amount, and specifications of the radioactive material proposed to be received and possessed at the geologic repository operations area...

(6) An identification of and justification for the selection of those variables, conditions, or other items which are determined to be probable subjects of license specifications. Special attention shall be given to those items that may significantly influence the final design...

Construction Authorization
60.31 Construction authorization.

Upon review and consideration of an application and environmental impact statement submitted under this part, the Commission may authorize construction if it determines:

(a) *Safety.* That there is reasonable assurance that the types and amounts of radioactive materials described in the application can be received, possessed, and disposed of in a geologic repository operations area of the design proposed without unreasonable risk to the health and safety of the public. In arriving at this determination, the Commission shall consider whether:

(1) DOE has described the proposed geologic repository including but not limited to:... (ii) the kinds and quantities of radioactive waste to be received, possessed, stored, and disposed of in the geologic repository operations area;.... (2) The site and

design comply with the performance objectives and criteria contained in Subpart E of this part...

Subpart E - Technical Criteria
Performance Objectives
60.111 Performance of the geologic repository operations area through permanent closure.

(a) *Protection against radiation exposures and releases of radioactive material.* The geologic repository operations area shall be designed so that until permanent closure has been completed, radiation exposures and radiation levels, and releases of radioactive materials to unrestricted areas, will be maintained within the limits specified in part 20 of this chapter and such generally applicable environmental standards for radioactivity as may have been established by Environmental Protection Agency.

Design Criteria for the Geologic Repository Operations Area
60.131 General design criteria for the geologic repository operations area.

(a) *Radiological protection.* The geologic repository operations area shall be designed to maintain radiation doses, levels, and concentrations of radioactive material in air in restricted areas within the limits specified in Part 20 of this chapter...

(b) *Protection against design basis events.* The structures, systems, and components important to safety shall be designed so that they will perform their necessary safety functions, assuming occurrence of design basis events.

(c) *Protection against dynamic effects of equipment failure and similar events.* The structures, systems, and components important to safety shall be designed to withstand dynamic effects such as missile impacts, that could result from equipment failure, and similar events and conditions that could lead to loss of their safety functions.

(d) *Protection against fires and explosions.*

(1) The structures, systems, and components important to safety shall be designed to perform their safety functions during and after credible fires or explosions in the geologic repository operations
area...

(3) The geologic repository operations area shall be designed to include explosion and fire detection alarm systems and appropriate suppression systems with sufficient capacity and capability to reduce the adverse effects of fires and explosions on structures, systems, and components important to safety.

(4) The geologic repository operations area shall be designed to include means to protect structures, systems, and components

important to safety against the adverse effects of either the operation or failure of the fire suppression systems...

(f) *Utility services.*

(1) Each utility service system that is important to safety shall be designed so that essential safety functions can be performed, assuming occurrence of the design basis events...

(i) *Instrumentation and control systems.* The design shall include provisions for instrumentation and control systems to monitor and control the behavior of systems important to safety, assuming occurrence of design basis events...

60.132 Additional design criteria for surface facilities in the geologic repository operations area.

(b) *Surface facility ventilation.* Surface facility ventilation systems supporting waste transfer, inspection, decontamination, processing, or packaging shall be designed to provide protection against radiation exposures and offsite releases as provided in § 60.111(a).

(c) *Radiation control and monitoring--(1) Effluent control.* The surface facilities shall be designed to control the release of radioactive materials in effluents during Category 1 design basis events so as to meet the performance objectives of § 60.111(a).

60.133 Additional design criteria for the underground facility.

(a) General criteria for the underground facility...

(2)The underground facility shall be designed so that the effects of credible disruptive events during the period of operations, such as flooding, fires and explosions, will not spread through the facility...

(e) Underground openings...

(1) Openings in the underground facility shall be designed so that operations can be carried out safely...

(2) Openings in the underground facility shall be designed to reduce the potential for deleterious rock movement or fracturing of overlying or surrounding rock.

(g) Underground facility ventilation. The ventilation system shall be designed to:

(1) Control the transport of radioactive particulates and gases within and releases from the underground facility in accordance with the performance objectives of § 60.111(a),

(2) Assure the ability to perform essential safety functions assuming occurrence of design basis events.

(3) Separate the ventilation of excavation and waste emplacement areas.

Preclosure Controlled Area
60.136 Preclosure controlled area.

(b) The geologic repository operations area shall be designed so that, for Category 2 design basis events, no individual located on or beyond any point on the boundary of the preclosure controlled area will receive the more limiting of a total effective dose equivalent of 0.05 Sv (5 rem), or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.5 Sv (50 rem). The eye dose equivalent shall not exceed 0.15 Sv (15 rem), and the shallow dose equivalent to skin shall not exceed 0.5 Sv (50 rem). The minimum distance from the surface facilities in the geologic repository operations area to the boundary of the preclosure controlled area must be at least 100 meters.

IV. ACCEPTANCE CRITERIA AND GUIDANCE

General Guidance

In general, the License Application for construction authorization must provide sufficient information to enable NRC to determine that there is reasonable assurance that the types and amounts of radioactive materials described in the License Application can be received, possessed, and disposed of in the Yucca Mountain geologic repository operations area of the design proposed without unreasonable risk to the health and safety of the public. To support that conclusion, this chapter of the License Application must:

- Identify potential design basis events for the repository.
- Categorize those events based on their probability of occurrence.
- Determine the radiological consequences[4], if any, of design basis events.
- Demonstrate that such radiological consequences are within prescribed regulatory requirements, both for repository workers and members of the general public. In some cases, this demonstration may require that particular mitigation features be incorporated into the repository design, or that specific administrative limits be adopted.

Chapter 7 must address all estimated doses, including those to the workers and to the public. The exception is that Chapter 10 contains the dose assessment for occupational dose for normal activities, which include infrequent activities, and identifies the source terms for occupational

[4]Applicable source terms are to be included in Chapter 9 (for radwaste management systems) and Chapter 10 (for the received spent nuclear fuel and high-level radioactive waste, and other sources, such as calibration and radiography sources) of the License Application.

exposures for direct exposure from the waste package and from other sources. Chapter 9 identifies the source terms for the radioactive waste management systems.

Most of the guidance in this chapter, other than that specified in 10 CFR 20 and 10 CFR 60, is based on and/or extracted from the following sources:

- NUREG-1323, *License Application Review Plan for a Geologic Repository for Spent Nuclear Fuel and High-Level Radioactive Waste* (NRC 1995a)
- YMP/97-01Q, *Management Plan for the Development of a Project Integrated Safety Assessment (PISA)* (YMP 1997)
- NUREG-1567, *Standard Review Plan for Spent Fuel Dry Storage Facilities* (NRC 1996c)
- NUREG-1536, *Standard Review Plan for Dry Cask Storage Systems* (NRC 1997a)
- *10 CFR Part 60, RIN 3150-AD51, Disposal of High-Level Radioactive Wastes in Geologic Repositories; Design Basis Events* (NRC 1996a)
- Regulatory Guide DG-3003, *Format and Content for the License Application for the High-Level Waste Repository* (NRC 1990)

NUREG-1323 (NRC 1995a) is largely incomplete and will become inapplicable to Yucca Mountain upon the expected issuance of a new 10 CFR 63 as discussed in Section III above. Information from NUREG-1323 (NRC 1995a) considered likely to be applicable under the new regulation is included in the technical guidance document, though the NUREG is not referenced as an authoritative source for such information. The *Management Plan for the Development of a Project Integrated Safety Assessment* (PISA) (YMP 1997) contains the Project Integrated Safety Assessment Content Guide that has been developed by the Yucca Mountain Site Characterization Project. The Project Integrated Safety Assessment Content Guide is the starting point for technical guidance document development. As is the case for NUREG-1323 (NRC 1995a), DG-3003 (NRC 1990) will become inapplicable to Yucca Mountain upon the expected issuance of a new 10 CFR 63 as discussed in Section III above. Information from DG-3003 (NRC 1990) considered likely to be applicable under the new regulation is included in the technical guidance document, though DG-3003 is not referenced as an authoritative source for such information. NUREG-1567 (NRC 1996c) and NUREG-1536 (NRC 1997a), though not directly applicable to the repository or the waste package, pertain to facilities and containers for dry storage of spent nuclear fuel and therefore contain useful and transferable information, including detailed information that is missing from the other NRC guidance documents for the repository. The rulemaking associated with revisions to 10 CFR 60, RIN 3150-AD51 (NRC 1996a), provides NRC guidance on the treatment of design basis events.

In some cases, the current state of repository design or program development may not be sufficient to provide the information requested. In those cases, a description of the criteria to be used will suffice. More detail on these topics will then be requested when the update to the License Application to receive and possess radioactive waste is prepared.

7.0 PURPOSE, SUMMARY, AND CONCLUSIONS

Provide the purpose for Chapter 7 of the License Application for construction authorization, and relate the information in other chapters that provide input to the safety analyses in Chapter 7. Provide a summary of the information contained in the chapter and a discussion of the organization of the chapter. Provide a summary of conclusions supporting the intent of this chapter and a description of important uncertainties, if applicable, that affect the conclusions.

7.1 REQUIREMENTS AND SCOPE

Describe the preclosure radiation exposure limits applicable to the proposed repository. Address how information contained in Chapter 7 relates to the eventual demonstration of compliance with the NRC requirements stated in 10 CFR 60. Discuss the inclusion, by reference, of the requirements in 10 CFR 20 and those requirements expected from the U.S. Environmental Protection Agency (EPA), as applicable. The requirements of interest in this chapter deal with preclosure radiological safety. As called out in 10 CFR 20 and 10 CFR 60, the potential exposures to members of the public and employees should be estimated for all representative design basis events. With the promulgation of the "design basis event amendment" to 10 CFR 60, effective January 3, 1997, NRC defines Categories 1 and 2 of design basis events as encompassing the full spectrum of events, including normal operations, anticipated off-normal operational events, and accidents.

Describe the scope of Chapter 7, which will be to demonstrate that the repository will comply with all applicable radiological requirements for the preclosure period as defined in 10 CFR 60. This evaluation of compliance will consist of design basis event identification, categorization, and evaluation against the defined applicable radiological limits. Information that is both design-specific and site-specific will be used in the evaluations. Indicate that some aspects of repository performance that are considered to have an impact on radiological safety will be addressed in other chapters. In particular, the as low as reasonably achievable philosophy, assessment of occupational doses, and organization of the health physics program will be addressed in detail in Chapter 10. Also, the demonstration of compliance with the criticality requirement during preclosure operations will be documented in Chapter 5, Section 5.4. The discussion in that section provides the basis for excluding inadvertent criticality as a design basis event.

7.2 APPROACH

Describe the approach used for assessing the radiological safety of the repository until permanent closure. It will be based on the design basis event approach outlined in 10 CFR 60 for items important to radiological controls and the therein mandated performance of the geologic repository operations area until permanent closure.

Organize the approach overview discussion such that it addresses the following six steps:

1. Identification of the conditions and events that could lead to radiation exposures, radiation levels, and releases of radioactive material to members of the public. The applicant is expected to use event trees, fault trees, and similar methods to identify

repository conditions potentially leading to radiological impacts on members of the public.

2. Estimation of the probabilities (numerical or qualitative) that these repository conditions and events may occur, or frequency of expected occurrence, and determination of the regulatory limits for impact of those probabilities (i.e., to categorize conditions or events as Category 1 or Category 2 design basis events). Per guidance provided in NRC 1996a: (a) identify Category 1 events as those having a frequency greater than or equal to $1x\ 10^{-2}$ per year; and (b) identify Category 2 events as those having a frequency less than $1x\ 10^{-2}$ per year but greater than or equal to $1x\ 10^{-6}$ per year. Events having a frequency less than $1x\ 10^{-6}$ per year are treated as beyond design basis events.

3. Analysis of 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 design basis events.

4. Identification and analysis of receptors (locations and lifestyles of people potentially exposed) for potential exposures and radiation levels, and releases of radioactive material as appropriate, respectively, for Category 1 and Category 2 design basis events.

5. Use of models to determine potential radiological impacts within the exposed population as appropriate for Category 1 and Category 2 design basis events. Radiological impact of all Category 1 events (including normal operations and likely events) are to be considered to determine the average annual exposures and releases. Radiological impact for Category 2 events are analyzed as consequences of each event as an independent occurrence.

6. Planning and design considerations used to meet 10 CFR 20 for workers for Category 1 design basis events.

For each step, describe where in the later sections of this chapter of the License Application more detailed information is provided. For topics outside the scope of this chapter of the License Application, such as definition of source terms, identify where in other License Application chapters the topic is addressed.

Ensure that the approach discussion describes the following:

- The extent to which a probabilistic risk assessment approach, as endorsed by NRC in (a) policy statements in NRC 1995b and NRC 1995c; (b) Regulatory Guide DG-1061, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis* (NRC, in draft); and (c) in rulemaking for 10 CFR 60 (NRC 1996a), is applied in assessing the performance of the repository.

- The extent to which a purely deterministic approach, as established in previous licensing of nuclear facilities, is used in assessing the performance of the repository; and the rationale for such augmentation.

- The extent to which the approach used parallels from the nuclear industry in developing the Safety Analysis Report and the License Applications for dry storage facilities or dry cask systems, insofar as that approach is understood by and has been previously accepted by NRC. Relate the discussion to the pertinent review plan elements in NUREG-1567 (NRC 1996c), Chapter 9, Radiation Protection Evaluation, and Chapter 12, Accident Analyses, and/or in NUREG-1536 (NRC 1997a). Identify where elements of the Yucca Mountain approach diverge from those in either or both NUREGs, and the bases for such diversions. Identify the inherent assumption that during preclosure operation, there exist administrative limits, procedures, training, and corrective and preventive maintenance programs that ensure that both the equipment and personnel will be available to respond to design basis events appropriately.

- The extent to which the approach used is consistent with the methodologies described in NUREG/CR-3332, *Radiological Assessment, a Textbook on Environmental Dose Analysis* (ORNL 1983).

- Elements of conservatism applied in the approach. For example, analyses of probabilities, exposures, and doses may be based on the following: (a) *best-estimate* parameters for Category 1 design basis events; and (b) *conservative* or *bounding* parameters for Category 2 design basis events.

- A discussion of beyond design basis events that address margins in the repository design to meeting the requirements of 10 CFR 60.

In each subsection in Section 7.2, begin by stating the purpose of the information in the section as it relates to the overall purpose of this chapter.

7.2.1 Event Type

Describe the methodology used to identify, screen for applicability, and categorize a comprehensive set of design basis events for the repository. Describe the preliminary hazard analysis process for initial identification and qualitative screening of candidate conditions and events.

Describe the two categories of design basis events as defined in 10 CFR 60. Category 1 is the set of design basis events likely to occur during the operational life of the repository. Category 2 is the set of design basis events unlikely to occur during the life of the facility but sufficiently credible to warrant consideration, taking into account the potential for significant radiological impacts on public health and safety. Cite NRC interpretations per NRC 1996a that design basis events comprise a sequence of events. Per guidance provided in NRC 1996a, identify the following: (a) Category 1 events as those having a frequency greater than or equal to 1×10^{-2} per year; and (b) Category 2 events as those having a frequency less than 1×10^{-2} per year but greater than or equal to 1×10^{-6} per year. Describe how events having a frequency less than 1×10^{-6} per

year are treated as beyond design basis events and will be used to demonstrate margins to compliance.

Describe how design basis events, other than "normal operations" design basis events, can be initiated by internal events, such as mechanical or other failures, operator error, or external events; that is, initiating events that would occur in the absence of the repository, such as earthquakes, tornadoes, and aircraft crashes. 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 design basis events 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.

Some incredible events (those with a probability below the lower limit for Category 2) may be included as design basis events for consideration of radiological consequences. If available, these analyses must be reported in Chapter 7 to demonstrate the limited hazard potential of operations being performed and/or the substantial defense-in-depth inherent in the design (i.e., the design has sufficient margin to protect the public even for very unlikely events, or the limiting design case).

Regulatory Guide 1.91, Rev. 1 (NRC 1978), may provide useful and applicable guidance for determining whether the risk of damage due to an explosion on a nearby transportation route is sufficiently high to warrant a detailed investigation. The project will assess the applicability of the guidance in that regulatory guide, and appropriate instructions for the License Application author will be provided in a future revision to this technical guidance document.

7.2.2 Radionuclide Releases

For the bounding event in each subset of design basis events, provide estimates of the types and quantities of radioactive releases that are calculated. Provide sufficient information to characterize the releases for each waste form. As appropriate, present comprehensive tables of the amount of material at risk in each unit of each waste form, the total number and annual rate of receipt and emplacement of each waste form, and the radioisotopic inventory in each waste form. (If such information is provided elsewhere in the report, so reference.) Show that, where possible, release assumptions with a licensing precedent are used. Types of radioactive releases are particles, volatile elements, and gases. Depending on the particular radionuclides involved, these types can change during the course of the release. Discuss temperature and location of the releases, along with other factors that can influence release fractions, if appropriate. 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 design basis events.

7.2.3 Radionuclide Transport and Radiation Dose

Describe the methodology used in later sections to predict the effects of radionuclide transport and the resultant radiological consequences to appropriate receptors. This discussion must cover the transport both of radionuclides within the facilities (hot cells, confinement barriers, filter systems) and outside the facility (atmospheric dispersion) to and beyond the boundaries of the facility. Also, discuss the techniques used for predicting ionizing radiation and radiological consequences.

Describe how the derived, radiological consequences for each bounding event will be compared with the mandated limits. The limits in 10 CFR 60.136 for members of the public only apply to Category 2 events. Where the radiological consequences exceed the applicable limits for a particular subset of design basis events, design modifications and/or administrative limits will be implemented to the extent required to ensure that the radiological consequences for that subset of design basis events remain less than the applicable limits. Describe the process whereby doses associated with each Category 1 event will be considered in demonstrating compliance. The limits in 10 CFR 20 for both repository workers and members of the public apply to Category 1 events. In addition, Category 1 events must comply with applicable environmental standards established by the EPA.

In addition, discuss the radionuclide pathways, calculational methodologies, and dose conversion factors used to arrive at dose predictions for workers and the public. Potential receptors will have been identified and used in calculational methods to ensure that compliance with 10 CFR 60 can be reasonably assured.

Summarize by describing how the approach described in Section 7.2 of the License Application demonstrates compliance with 10 CFR 60.21(c)(3)(i), 10 CFR 60.111(a), 10 CFR 60.131(a), 10 CFR 60.21(c)(3)(ii), and 10 CFR 60.136(b).

7.3 REPOSITORY DESIGN INPUT TO SAFETY ASSESSMENT

The vast majority of the information for this section will be included in other chapters in the License Application and can be incorporated by reference. For the preclosure radiological assessment, provide a simplified overview and a description of the structures, systems, and components from the surface and subsurface facilities and the engineered barrier system credited with preventing or mitigating a design basis event. Include salient details of the prevention and mitigation systems for those systems for which the design has been developed. Examples of such systems may include heating, ventilation, air conditioning, emergency power generation, monitoring systems, and emergency response systems. Most of this information will have been presented in detail in other chapters. Reference those chapters, including section/subsection numbers as appropriate.

In each subsection in Section 7.3, begin by stating the purpose of the information as it relates to the overall purpose of this chapter.

7.3.1 Repository Description

7.3.1.1 Surface Facilities

Provide an overview of salient features of surface facilities used for radiological safety, such as heating, ventilation, air conditioning, emergency power generation, monitoring systems, and emergency shutdown systems, that may provide some degree of mitigation and/or prevention. Incorporate details by reference to other appropriate chapters and sections of the License Application.

7.3.1.2 Subsurface Facilities

Provide an overview of salient features of subsurface facilities used for radiological safety. Incorporate details by reference to other appropriate chapters and sections of the License Application. Describe how Section 7.3.1.2, along with other chapters and sections of the License Application, demonstrates compliance with 10 CFR 60.133(a)(2), 10 CFR 60.133(g)(1), and 10 CFR 60.133(g)(2).

7.3.2 Waste Package Description

Provide an overview of salient features of the waste package with respect to radiological safety. Incorporate details by reference to other appropriate chapters and sections of the License Application. Discuss only information relevant to the integrity of the container. As appropriate, describe the relevant differences between waste packages used for the variety of waste forms. Summarize the discussion contained in Chapter 5, Section 5.4. The discussion in that section provides the basis for excluding inadvertent criticality as a design basis event. No criticality-related radiological safety issues need to be addressed in Chapter 7 of the License Application.

7.4 SOURCE TERM DEVELOPMENT

Describe the waste forms and their characteristics that affect preclosure radiological safety. Reference the other chapters and sections of the License Application containing detailed information, as appropriate. Describe how Section 7.2, along with other chapters and sections of the License Application, demonstrate compliance with 10 CFR 60.21(c)(5) and 10 CFR 60.31(a)(1)(ii).

In each subsection in Section 7.4, begin 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 the form, quantities, and physical characteristics of the spent nuclear fuel. Describe the estimated radionuclide inventory to the extent that it affects release, transport, or dose. Make appropriate reference to the source term information contained in Chapter 10 and the detailed waste descriptions in Chapter 5 of the License Application. Include a discussion of the characteristics of gaseous elements, volatile elements, and elements in solid form (refractory), including activated corrosion/wear products from reactor systems and the applicable radiation

shielding source term data (i.e., emission rates and energy spectra). Describe how spent nuclear fuel from various types of light water reactors are grouped according to similar physical characteristics and how bounding source terms are developed for each group.

7.4.2 High-Level Radioactive Waste

Describe the form, quantities, and physical characteristics of the waste. Describe the estimated radionuclide inventory to the extent that it affects release, transport, or dose. Make appropriate reference to the source term information contained in Chapter 10 and the detailed waste description in Chapter 5 of the License Application. 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 are grouped according to similar physical characteristics and how bounding source terms are developed for each group.

7.4.3 Radioactive Effluents and Wastes

Describe the form, quantities, and physical characteristics of the effluents and waste. Describe the estimated radionuclide inventory to the extent that it affects release, transport, or dose. Make appropriate reference to the source term information contained in Chapter 9. Any release of radioactive material during normal facility operation is, as defined in 10 CFR 60, a Category 1 design basis event, and, as such, must be included in design basis event safety assessments.

7.4.4 Other U.S. Department of Energy Spent Nuclear Fuel (if needed)

Discuss the salient features of the other DOE spent nuclear fuel waste forms, including a description of these waste forms for the parameters given in Section 7.4.1. Make reference to the detailed waste description in Chapter 5 of the License Application. Describe how DOE spent nuclear fuel from various sources are grouped according to similar physical characteristics and how bounding source terms are developed for each group.

7.4.5 Plutonium Disposition (if needed)

Discuss the salient features of the other DOE plutonium waste forms, including a description of these waste forms for the parameters given in Section 7.4.1. Make reference to the detailed waste description in Chapter 5 of the License Application.

7.5 DESIGN BASIS EVENT DEFINITION

In each subsection that follows in Section 7.5, begin 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 design basis events considered, and a general description of the processes used to screen and select the set of candidate events for further consideration and evaluation. Describe in detail the systematic review of all events and the data used to categorize

them as Category 1 or 2. Include the rationale for excluding any candidate event from the set considered.

7.5.2 Category 1 Design Basis Events

Provide summary detail of the events and event sequences determined to be Category 1 design basis events; that is, "Those natural and human induced events that are reasonably likely to occur regularly, moderately frequently, or one or more times before permanent closure of the geologic repository operations area" (10 CFR 60.2(1)(i)). To aid discussion, segregate the "normal operations" events from the other Category 1 design basis events (i.e., anticipated off-normal events). For the latter, describe how those design basis events of a common nature are placed into groups, and how, for each group, a bounding event of consequence is identified.

7.5.3 Category 2 Design Basis Events

Provide summary detail of those events defined as Category 2 design basis events (unlikely but credible events that are not expected to occur during the preclosure period). Describe how design basis events of a similar nature are identified and how the bounding event for each class is defined. If results of consequence analyses for nonmechanistic failures of containment barrier systems are presented, compliance can be based on offsite dose limits for Category 2, per 10 CFR 60.136.

7.6 POTENTIAL RELEASES FROM DESIGN BASIS EVENTS

In each subsection that follows in Section 7.6, begin by stating the purpose of the information as it relates to the overall purpose of this chapter.

7.6.1 Initiating Events and Accident Sequence

Describe the features, events, and processes for each design basis event that are potential contributors to radiological releases to the environment and/or exposures to the public or occupational workers. Include internal events that are failures within the operating plant, such as a dropped waste package, rock falls, or dropped bare fuel assemblies. Include discussion of external events, such as earthquake, flood, fire, wind-driven missile, and aircraft crash (depending on frequency). Summarize the results of mechanical response calculations of the waste form and its container to selected bounding design basis events.

7.6.2 Repository Radionuclide Transport Mechanisms

Address the generation of the radionuclide source term from the waste form and the container. Discuss the transportation of radionuclides released from the waste containment barriers to the outside environment. Include particulate matter, vapor phases, and fission product gases in the source term. For particulate transport, include a discussion of any mitigating features, events, and processes, such as fallout, plateout, and washout, in the various consequence evaluations. Include effects of the ventilation system, including operation of the radiation protection system and high-efficiency particulate air filter system.

7.7 ATMOSPHERIC DISPERSION AND DOSE CALCULATIONS

Describe the analysis of the transportation of the radionuclides from point of release to the boundary of the preclosure controlled area and distances beyond, as appropriate, to assess offsite dose consequences for preclosure events. 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 and methods and established dose conversion factors.

Describe the analysis of occupational dose to repository workers for Category 1 design basis events other than the "normal operations" design basis event. The analysis of occupational dose to repository workers for Category 2 design basis events is not required in accordance with 10 CFR 60.

In each subsection that follows in Section 7.7, begin by stating the purpose of the information as it relates to the overall purpose of the chapter.

The following regulatory guides may provide useful and applicable guidance for methodologies and/or assumptions to be used to calculate radionuclide dispersions and radiological consequences. The project will assess the applicability of the guidance in these regulatory guides and appropriate instructions for the License Application author will be provided in a future revision to this technical guidance document.

- Regulatory Guide 1.3, *Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors* (NRC 1974a)

- Regulatory Guide 1.4, *Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors* (NRC 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* (NRC 1972)

- Regulatory Guide 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants Accident Analysis* (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 1977b)

- 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 1997b)

- 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 1977a)

- 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, making reference to Section 3.4, as appropriate. Describe how the data are reduced to generate the required values of atmospheric dispersion coefficients. Any special atmospheric dispersion conditions applying to the site and having radiological repercussions shall be identified and an assessment made for impact on dose predictions. Regulatory Guides 1.111 (NRC 1977b) and 1.145 (NRC 1982) may provide useful and applicable guidance on the methods to be used to calculate radionuclide dispersion.

7.7.2 Radiological Dose Estimates

Derive and discuss radiological consequences for the events considered. The process of calculating doses to people, after the radionuclide dispersion has been calculated, includes: selection of the exposure scenario (critical group) in accordance with regulatory guidance, selection of the important pathways, modeling of environmental transport of radionuclides, determination of radionuclide intakes, and doses resulting from these intakes.

For Category 1 design basis events, describe how the annual doses to both workers and members of the public are derived. Regulatory Guide 1.109 (NRC 1977a) may provide useful and applicable guidance on the methods to be used to calculate 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 as low as reasonably achievable issues where appropriate.

For Category 2 design basis events, only the doses to the members of the public at the preclosure controlled area boundary are calculated. Regulatory Guides 1.3 (NRC 1974a), 1.4 (NRC 1974b), and 1.25 (NRC 1972) may provide useful and applicable guidance on the methods to be used to calculate 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 in 10 CFR 60.

Multiple pathways shall be considered in these dose estimates. This section shall include a discussion of total effective dose equivalent. This section also shall include 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 to be used to calculate occupational doses. The project will assess the applicability of the guidance in this regulatory guide, and appropriate instructions for the License Application author will be provided in a future revision to this technical guidance document.

Show compliance with 10 CFR 60.21(c)(3)(i), 10 CFR 60.111(a), 10 CFR 60.131(a), 10 CFR 60.21(c)(3)(ii), and 10 CFR 60.136(b). For Category 1 design basis events, each of the radiation dose limits in 10 CFR 20 shall be addressed and compared with the regulatory standard. For Category 2 design basis events, expected doses shall be compared with the criteria in 10 CFR 60. Describe any instances where mitigating features and/or administrative limits are required to maintain the radiological consequences of particular design basis events or groups of similar design basis events within the applicable limits.

7.8 CONCLUSIONS

Describe how Chapter 7 of the License Application demonstrates compliance, either wholly or in conjunction with other chapters and sections of the License Application, as appropriate, with 10 CFR 60.21(c)(3)(i) and (ii); 10 CFR 60.31(a) and (a)(2); 10 CFR 60.111(a); 10 CFR 60.131(a), (b), (c), (d)(1), (d)(3), (d)(4), (f)(1), and (i); 10 CFR 60.132(b) and (c)(1); 10 CFR 60.133(a)(2), (g)(1), and (g)(2); and 10 CFR 60.136(b).

7.8.1 Category 1 Design Basis Events - Demonstration of Compliance to 10 CFR 60.111(a)

Describe the radiological consequences due to Category 1 design basis events during the operation of the repository until permanent closure. Apply the methods described in Section 7.2.3 to consider all Category 1 events, including off-normal and normal events. Provide a summary table showing the calculated dose for each off-normal event and the estimated annual probability of the event. Compare the radiological consequences with the applicable limits of 10 CFR 60.111(a) and demonstrate compliance thereto. In particular, demonstrate compliance with limits of 10 CFR 20 and to applicable environmental standards established by the EPA. Describe instances where mitigating features and/or administrative limits are required to show protection for particular design basis events. Conclude that the requirements of 10 CFR 60.111(a) will be met, assuming occurrence of Category 1 design basis events.

7.8.2 Category 2 Design Basis Events - Demonstration of Compliance to 10 CFR 60.136(b)

Describe the radiological consequences due to Category 2 design basis events during the operation of the repository until permanent closure. Compare the predicted radiological consequences with the applicable limits. Describe instances where mitigating features and/or administrative limits are required to show protection for particular design basis events. Conclude that the requirements of 10 CFR 60.136 will be met, assuming occurrence of Category 2 design basis events.

V. REFERENCES

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

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

10 CFR 60. Energy: Disposal of High-Level Radioactive Wastes in Geologic Repositories.

NRC (U.S. Nuclear Regulatory 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. Government Printing Office. 223044.

NRC 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. Government Printing Office. 4652.

NRC 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. Government Printing Office. 4651.

NRC 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. Government Printing Office. 222664.

NRC 1977a. *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. Government Printing Office. 223058.

NRC 1977b. *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. Government Printing Office. 2780.

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. Government Printing Office. 2774.

NRC 1982. *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants Accident Analysis.* Regulatory Guide 1.145. Rev. 1. Washington, D.C.: U.S. Government Printing Office. 2791, 222640.

NRC 1990. *Format and Content for the License Application for the High-Level Waste Repository.* Regulatory Guide DG-3003. Washington, D.C.: U.S. Government Printing Office. 223107.

NRC 1992. *Monitoring Criteria and Methods to Calculate Occupational Radiation Doses.* Regulatory Guide 8.34. Rev. 0. Washington, D.C.: U.S. Government Printing Office. 8912, 237835.

NRC 1995a. *License Application Review Plan for a Geologic Repository for Spent Nuclear Fuel and High-Level Radioactive Waste.* NUREG-1323. Rev. 1. Washington, D.C.: U.S. Government Printing Office. 213043.

NRC 1995b. *Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement.* Federal Register, Volume 60, p.42622, August 16, 1995. Document on order.

NRC 1995c. *Framework for Applying Probabilistic Risk Analysis in Reactor Regulation.* SECY-95-280, November 27, 1995. 232400.

NRC 1996a. *10 CFR Part 60, RIN 3150-AD51, Disposal of High-Level Radioactive Wastes in Geologic Repositories; Design Basis Events.* Federal Register, Volume 61, Number 234,p. 64257-64270, December 4, 1996. Document on order.

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. Government Printing Office. 229689.

NRC 1996c. *Standard Review Plan for Spent Fuel Dry Storage Facilities.* NUREG-1567. Washington, D.C.: U.S. Government Printing Office. 226657.

NRC 1997a. *Standard Review Plan for Dry Cask Storage Systems.* NUREG-1536. Washington, D.C.: U.S. Government Printing Office. 226847.

NRC 1997b. *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. Government Printing Office. 222663.

ORNL (Oak Ridge National Laboratory) 1983. *Radiological Assessment, a Textbook on Environmental Dose Analysis.* NUREG/CR-3332. Oak Ridge, Tennessee. 223809.

YMP (Yucca Mountain Site Characterization Project) 1997. *Management Plan for the Development of a Project Integrated Safety Assessment (PISA).* YMP/97-01Q. Rev. 2. Las Vegas, Nevada: Yucca Mountain Site Characterization Office. MOL.19971017.0200.

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CHAPTER 8. PERFORMANCE OF THE REPOSITORY AFTER PERMANENT CLOSURE

I. OBJECTIVE

This chapter provides guidance for the author(s) of Chapter 8 of the License Application. The License Application will be submitted to the U.S. Nuclear Regulatory Commission (NRC) for authorization to construct a geologic high-level radioactive waste repository at Yucca Mountain. This chapter discusses the performance assessment related information needed to support the License Application. This chapter evaluates the performance of the entire repository system, including both the engineered and natural system components.

The author must determine whether the release of radioactive materials to the accessible environment can meet the applicable standards as established by U.S. Environmental Protection Agency (EPA) and NRC implementation regulations. The relationship between information presented in this chapter and the NRC key technical issues (KTIs) must be addressed. Rationale and justification of all input and methodologies used in compliance assessment for the License Application must be presented. Review the contents of Chapter 6 of the *Mined Geologic Disposal System License Application Annotated Outline,* YMP/94-05 (DOE Office of Civilian Radioactive Waste Management, in draft), for additional guidance on total system performance assessment.

II. GUIDANCE TOPICS

The following topics will be discussed in Chapter 8 of the License Application:

- 8.0 PURPOSE, SUMMARY, AND CONCLUSIONS
- 8.1 INTRODUCTION
  - 8.1.1 Purpose
  - 8.1.2 Scope
  - 8.1.3 Basic Approach
- 8.2 SYSTEM AND SUBSYSTEM DESCRIPTIONS
  - 8.2.1 Site Description and Models
  - 8.2.2 Repository Description and Models
  - 8.2.3 Waste Package/Engineered Barrier System Description and Models
  - 8.2.4 Biosphere Models
  - 8.2.5 Description of Potentially Disruptive Features, Events, and Processes, and Their Corresponding Models
- 8.3 EVALUATION OF UNDISTURBED PERFORMANCE
  - 8.3.1 Waste Package/Engineered Barrier System Performance
  - 8.3.2 Natural Barrier System Performance
  - 8.3.3 Sensitivity of Undisturbed Performance Results
- 8.4 EVALUATION OF POTENTIALLY DISTURBED PERFORMANCE
  - 8.4.1 Identification of Features, Events, and Processes

8.4.2 Screening of Features, Events, and Processes
8.4.3 Evaluation of Disturbed Scenario Probabilities
8.5 CONCLUSIONS

III. REGULATORY REQUIREMENTS

***** IMPORTANT NOTE *****

At the time of development of this document, it appears likely that NRC will issue a new, site-specific regulation for Yucca Mountain in 1999. At the time the new 10 CFR 63 is issued, 10 CFR 60 will be modified to state that it is not applicable to Yucca Mountain. Details about the new regulation are not known. However, it has been determined prudent to develop the working draft License Application in accordance with the best available estimate of the form of the new regulation. To avoid confusing references and to allow production of the technical guidance document in time to support training for the working draft License Application, the text of the technical guidance document that follows this note has not been revised to reflect the postulated form of the new regulation. Instead, this note provides guidance needed by working draft License Application authors to comply with management direction on addressing the new regulation. The paragraphs that follow explain changes in the regulation with respect to the existing 10 CFR 60 that affects this technical guidance document chapter, and provide guidance on how to address the changes in the working draft License Application. Where guidance in this note conflicts with guidance in the remainder of this chapter of the technical guidance document, the guidance in the note is to govern. Any questions should be addressed to Management and Operating Contractor (M&O) Licensing.

Changes to Regulations Applicable to Chapter 8

The following 10 CFR 60 requirements applicable to technical guidance document Chapter 8 and listed below in Section III are assumed to not be carried over to the new regulation: 60.113 and 60.133.

In place of these specific requirements, the new regulation is expected to require a discussion of the contribution of the individual barriers to the overall performance of the repository. The manner in which that contribution is to be described has not been specified, though an NRC review plan on the subject may be issued sometime in the future.

Guidance to Working Draft License Application Authors for Addressing Changes

- Disregard guidance in Section IV regarding demonstrating compliance with any of the regulations assumed to be deleted as described above. Instead, discuss the performance of the engineered barrier system and its components in supporting overall repository performance.

- Address the technical issue of each of the regulations in Section III, including those expected to be deleted, but do not relate the discussion to demonstrating compliance with any of the deleted regulations. Continue to explicitly address the regulations in Section III that are not expected to be deleted.

- Detailed discussion of design criteria, design bases, and performance is still required as per the existing guidance in Section IV. Discussion should still include detailed discussions of the phenomena that affect that performance.

- No sections or subsections of Chapter 8 are expected to be deleted as a result of this note.

***** END OF NOTE *****

Total system performance standards are included in 10 CFR 60, in which NRC references environmental (radiological) health standards issued by EPA. For many years there has been uncertainty in the total system performance standard. Title 40 CFR 191, as promulgated in 1985, was used as one measure of total system performance. However, this standard was remanded to EPA for further consideration and, as subsequently reissued, no longer applies to the Yucca Mountain repository. The Energy Policy Act of 1992 (EPAct 1992) directed EPA to promulgate a standard specifically for a repository at Yucca Mountain based on and consistent with the recommendations of the National Academy of Sciences (NAS 1995). EPA is currently drafting this standard. NRC will then revise 10 CFR 60 to be consistent with the EPA standard. Appendix A provides a cross-reference between the regulations in 10 CFR 60 and the specific License Application sections that address these regulations.

Sections of the regulations that are to be addressed fully or in part in Chapter 8 of the License Application are reproduced below. These are verbatim extracts from the regulations. Paragraph numbers refer to Title 10 of the U.S. Code of Federal Regulations (CFR). Revisions to regulations are not uncommon. Verify that current versions are used.

> **10 CFR Part 60, Disposal of High-Level Radioactive Wastes in Geologic Repositories**
> **10 CFR Part 60.2 Definitions.**
> As used in this part:
>
> Anticipated processes and events means those natural processes and events that are reasonably likely to occur during the period the intended performance objective must be achieved. To the extent reasonable in the light of the geologic record, it shall be assumed that those processes operating in the geologic setting during the Quaternary Period continue to operate but with the perturbations caused by the presence of emplaced radioactive waste superimposed thereon.
>
> Unanticipated processes and events means those processes and events affecting the geologic setting that are judged not to be reasonably likely to occur during the period the intended performance objective must be achieved, but which are

nevertheless sufficiently credible to warrant consideration. Unanticipated processes and events may be either natural processes or events or processes and events initiated by human activities other than those activities licensed under this part. Processes and events initiated by human activities may only be found to be sufficiently credible to warrant consideration if it is assumed that: (1) the monuments provided for by this part are sufficiently permanent to serve their intended purpose; (2) the value to future generations of potential resources within the site can be assessed adequately under the applicable provisions of this part; (3) an understanding of the nature of radioactivity, and an appreciation of its hazards, have been retained in some functioning institutions; (4) institutions are able to assess risk and to take remedial action at a level of social organization and technological competence equivalent to, or superior to, that which was applied in initiating the processes or events concerned; and (5) relevant records are preserved, and remain accessible, for several hundred years after permanent closure.

Accessible environment means:

1. The atmosphere;
2. The land surface;
3. Surface water;
4. Oceans; and
5. The portion of the lithosphere that is outside the postclosure controlled area.

60.21 Content of Application.

(a) An application shall consist of general information and a Safety Analysis Report.

(c) The Safety Analysis Report shall include:

(1) A description and assessment of the site at which the geologic repository operations area is to be located with appropriate attention to those features of the site that might affect geologic repository operations area design and performance. The description of the site shall identify the location of the geologic repository operations area with respect to the boundary of the accessible environment.

(ii) The assessment shall contain:

(B) Analyses to determine the degree to which each of the favorable and potentially adverse conditions, if present, has been characterized, and the extent to which it contributes to or detracts

from isolation. For the purpose of determining the presence of the potentially adverse conditions, investigations shall extend from the surface to a depth sufficient to determine critical pathways for radionuclide migration from the underground facility to the accessible environment. Potentially adverse conditions shall be investigated outside of the postclosure controlled area if they affect isolation within the postclosure controlled area.

(c) An evaluation of the performance of the geologic repository for the period after permanent closure, assuming anticipated processes and events, giving the rates and quantities of releases of radionuclides to the accessible environment as a function of time; and a similar evaluation which assumes the occurrence of unanticipated processes and events.

(D) The effectiveness of engineered and natural barriers, including barriers that may not be themselves a part of the geologic repository operations area, against the release of radioactive material to the environment. The analysis shall also include a comparative evaluation of alternatives to the major design features that are important to waste isolation, with particular attention to the alternatives that would provide longer radionuclide containment and isolation.

(F) An explanation of measures used to support the models used to perform the assessments required in paragraphs (A) through (D). Analyses and models that will be used to predict future conditions and changes in the geologic setting shall be supported by using an appropriate combination of such methods as field tests, in situ tests, laboratory tests which are representative of field conditions, monitoring data, and natural analog studies.

60.31 Construction authorization.

Upon review and consideration of an application and environmental impact statement submitted under this part, the Commission may authorize construction if it determines:

(a) Safety. That there is reasonable assurance that the types and amounts of radioactive materials described in the application can be received, possessed, and disposed of in a geologic repository operations area of the design proposed without unreasonable risk to the health and safety of the public. In arriving at this determination, the Commission shall consider whether:

(2) The site and design comply with the performance objectives and criteria contained in Subpart E of this part. Subpart E contain the Technical Criteria:

§ 60.122 Siting criteria.

§ 60.131 General design criteria for the geologic repository operations area.

§ 60.132 Additional design criteria for surface facilities in the geologic repository operations area.

§ 60.133 Additional design criteria for the underground facility.

§ 60.134 Design of seals for shafts and boreholes.

§ 60.135 Criteria for the waste package and its components.

60.101 Purpose and nature of findings.

(a)(1) Subpart B of this part prescribes the standards for issuance of a license to receive and possess source, special nuclear, or byproduct material at a geologic repository operations area. In particular, § 60.418 requires a finding that the issuance of a license will not constitute an unreasonable risk to the health and safety of the public. The purpose of this subpart is to set out performance objectives and site and design criteria which, if satisfied, will support such a finding of no unreasonable risk.

(2) While these performance objectives and criteria are generally stated in unqualified terms, it is not expected that complete assurance that they will be met can be presented. A reasonable assurance, on the basis of the record before the Commission, that the objectives and criteria will be met is the general standard that is required. For § 60.112, and other portions of this subpart that impose objectives and criteria for repository performance over long times into the future, there will inevitably be greater uncertainties. Proof of the future performance of engineered barrier systems and the geologic setting over time periods of many hundreds or many thousands of years is not to be had in the ordinary sense of the word. For such long-term objectives and criteria, what is required is reasonable assurance, making allowance for the time period, hazards, and uncertainties involved, that the outcome will be in conformance with those objectives and criteria. Demonstration of compliance with such objectives and criteria will involve the use of data from accelerated tests and predictive models that are supported by such measures as field and laboratory tests, monitoring data and natural analog studies.

(b) Subpart B of this part also lists findings that must be made in support of an authorization to construct a geologic repository operations area. In particular, § 60.31(a) requires a finding that there is reasonable assurance that the types and amounts of radioactive materials described in the application can be received, possessed, and disposed of in a geologic repository operations area of the design proposed without unreasonable risk to the health and safety of the public. As stated in that paragraph, in arriving at this determination, the Commission will consider whether the site and design comply with the criteria contained in this subpart. Once again, while the criteria may be written in unqualified terms, the

demonstration of compliance may take uncertainties and gaps in knowledge into account, provided that the Commission can make the specified finding of reasonable assurance as specified in paragraph (a) of this section.

60.102 Concepts.

This section provides a functional overview of subpart E.

In the event of any inconsistency with definitions found in § 60.2, those definitions shall prevail.

(e) Isolation of waste.

(1) During the first several hundred years following permanent closure of a geologic repository, when radiation and thermal levels are high and the uncertainties in assessing repository performance are large, special emphasis is placed upon the ability to contain the wastes by waste packages within an engineered barrier system. This is known as the containment period. The engineered barrier system includes the waste packages and the underground facility. A waste package is composed of the waste form and any containers, shielding, packing, and absorbent materials immediately surrounding an individual waste container. The underground facility means the underground structure, including openings and backfill materials, but excluding, shafts, boreholes, and their seals.

(2) Following the containment period special emphasis is placed upon the ability to achieve isolation of the wastes by virtue of the characteristics of the geologic repository. The engineered barrier system works to control the release of radioactive material to the geologic setting and the geologic setting works to control the release of radioactive material to the accessible environment. Isolation means inhibiting the transport of radioactive material so that amounts and concentrations of the materials entering the accessible environment will be kept within prescribed limits.

60.112 Overall system performance objective for the geologic repository after permanent closure.

The geologic setting shall be selected and the engineered barrier system and the shafts, boreholes and their seals shall be designed to assure that releases of radioactive materials to the accessible environment following permanent closure conform to such generally applicable environmental standards for radioactivity as may have been established by the Environmental Protection Agency with respect to both anticipated processes and events and unanticipated processes and events.

60.113 Performance of particular barriers after permanent closure.

(a) General provisions-(1) Engineered barrier system.

(i) The engineered barrier system shall be designed so that assuming anticipated processes and events:

(A) Containment of HLW will be substantially complete during the period when radiation and thermal conditions in the engineered barrier system are dominated by fission product decay; and

(B) any release of radionuclides from the engineered barrier system shall be a gradual process which results in small fractional releases to the geologic setting over long times. For disposal in the saturated zone, both the partial and complete filling with groundwater of available void spaces in the underground facility shall be appropriately considered and analyzed among the anticipated processes and events in designing the engineered barrier system.

(ii) In satisfying the preceding requirement, the engineered barrier system shall be designed, assuming anticipated processes and events, so that:

(A) Containment of HLW within the waste packages will be substantially complete for a period to be determined by the Commission taking into account the factors specified in § 60.113(b) provided, that such period shall be not less than 300 years nor more than 1,000 years after permanent closure of the geologic repository; and

(B) The release rate of any radionuclide from the engineered barrier system following the containment period shall not exceed one part in 100,000 per year of the inventory of that radionuclide calculated to be present at 1,000 years following permanent closure, or such other fraction of the inventory as may be approved or specified by the Commission; provided, that this requirement does not apply to any radionuclide which is released at a rate less than 0.1% of the calculated total release rate limit. The calculated total release rate limit shall be taken to be one part in 100,000 per year of the inventory of radioactive waste, originally emplaced in the underground facility, that remains after 1,000 years of radioactive decay.

(2) Geologic setting. The geologic repository shall be located so that pre-waste-emplacement groundwater travel time along the fastest path of likely radionuclide travel from the disturbed zone to the accessible environment shall be at least 1,000 years or such other travel time as may be approved or specified by the Commission.

(b) On a case-by-case basis, the Commission may approve or specify some other radionuclide release rate, designed containment period or pre-waste-emplacement groundwater travel time,

provided that the overall system performance objective, as it relates to anticipated processes and events, is satisfied. Among the factors that the Commission may take into account are:

(1) Any generally applicable environmental standard for radioactivity established by the Environmental Protection Agency;

(2) The age and nature of the waste, and the design of the underground facility, particularly as these factors bear upon the time during which the thermal pulse is dominated by the decay heat from the fission products;

(3) The geochemical characteristics of the host rock, surrounding strata and groundwater; and

(4) Particular sources of uncertainty in predicting the performance of the geologic repository.

(c) Additional requirements may be found to be necessary to satisfy the overall system performance objective as it relates to unanticipated processes and events.

60.122 Siting criteria.

(a)(1) A geologic setting shall exhibit an appropriate combination of the conditions specified in paragraph (b) of this section so that, together with the engineered barriers system, the favorable conditions present are sufficient to provide reasonable assurance that the performance objectives relating to isolation of the waste will be met.

(2) If any of the potentially adverse conditions specified in paragraph (c) of this section is present, it may compromise the ability of the geologic repository to meet the performance objectives relating to isolation of the waste. In order to show that a potentially adverse condition does not so compromise the performance of the geologic repository the following must be demonstrated:

(ii) The effect of the potentially adverse human activity or natural condition on the site has been adequately evaluated using analyses which are sensitive to the potentially adverse human activity or natural condition and assumptions which are not likely to underestimate its effect; and

(iii)(A) The potentially adverse human activity or natural condition is shown by analysis pursuant to paragraph (a)(2)(ii) of this section not to affect significantly the ability of the geologic repository to meet the performance objectives relating to isolation of the waste, or

(B) The effect of the potentially adverse human activity or natural condition is compensated by the presence of a combination of the favorable characteristics so that the performance objectives relating to isolation of the waste are met.

(b) Favorable conditions. (1) The nature and rates of tectonic, hydrogeologic, geochemical, and geomorphic processes (or any of such processes) operating within the geologic setting during the Quaternary Period, when projected, would not affect or would favorably affect the ability of the geologic repository to isolate the waste.

(3) Geochemical conditions that:

(i) Promote precipitation or sorption of radionuclides;

(ii) Inhibit the formation of particulates, colloids, and inorganic and organic complexes that increase the mobility of radionuclides; or

(iii) Inhibit the transport of radionuclides by particulates, colloids, and complexes.

(4) Mineral assemblages that, when subjected to anticipated thermal loading, will remain unaltered or alter to mineral assemblages having equal or increased capacity to inhibit radionuclide migration.

(5) Conditions that permit the emplacement of waste at a minimum depth of 300 meters from the ground surface. (The ground surface shall be deemed to be the elevation of the lowest point on the surface above the disturbed zone.)

(6) A low population density within the geologic setting and a postclosure controlled area that is remote from population centers.

(7) Pre-waste-emplacement groundwater travel time along the fastest path of likely radionuclide travel from the disturbed zone to the accessible environment that substantially exceeds 1,000 years.

(8) For disposal in the unsaturated zone, hydrogeologic conditions that provide-

(i) Low moisture flux in the host rock and in the overlying and underlying hydrogeologic units;

(ii) A water table sufficiently below the underground facility such that fully saturated voids contiguous with the water table do not encounter the underground facility;

(iii) A laterally extensive low-permeability hydrogeologic unit above the host rock that would inhibit the downward movement of water or divert downward moving water to a location beyond the limits of the underground facility;

(iv) A host rock that provides for free drainage; or (v) A climatic regime in which the average annual historic precipitation is a small percentage of the average annual potential evapotranspiration.

8 Potentially adverse conditions. The following conditions are potentially adverse conditions if they are characteristic of the controlled area or may affect isolation within the postclosure controlled area.

(1) Potential for flooding of the underground facility, whether resulting from the occupancy and modification of floodplains or from the failure of existing or planned man-made surface water impoundments.

(2) Potential for foreseeable human activity to adversely affect the groundwater flow system, such as groundwater withdrawal, extensive irrigation, subsurface injection of fluids, underground pumped storage, military activity or construction of large scale surface water impoundments.

(3) Potential for natural phenomena such as landslides, subsidence, or volcanic activity of such a magnitude that large-scale surface water impoundments could be created that could change the regional groundwater flow system and thereby adversely affect the performance of the geologic repository.

(4) Structural deformation, such as uplift, subsidence, folding, or faulting that may adversely affect the regional groundwater flow system.

(5) Potential for changes in hydrologic conditions that would affect the migration of radionuclides to the accessible environment, such as changes in hydraulic gradient, average interstitial velocity, storage coefficient, hydraulic conductivity, natural recharge, potentiometric levels, and discharge points.

(6) Potential for changes in hydrologic conditions resulting from reasonably foreseeable climatic changes.

(7) Groundwater conditions in the host rock, including chemical composition, high ionic strength or ranges of Eh-pH, that could increase the solubility or chemical reactivity of the engineered barrier system.

(8) Geochemical processes that would reduce sorption of radionuclides, result in degradation of the rock strength, or adversely affect the performance of the engineered barrier system.

(9) Groundwater conditions in the host rock that are not reducing.

(10) Evidence of dissolutioning such as breccia pipes, dissolution cavities, or brine pockets.

(11) Structural deformation such as uplift, subsidence, folding, and faulting during the Quaternary Period.

(12) Earthquakes which have occurred historically that if they were to be repeated could affect the site significantly.

(13) Indications, based on correlations of earthquakes with tectonic processes and features, that either the frequency of occurrence or magnitude of earthquakes may increase.

(14) More frequent occurrence of earthquakes or earthquakes of higher magnitude than is typical of the area in which the geologic setting is located.

(15) Evidence of igneous activity since the start of the Quaternary Period.

(16) Evidence of extreme erosion during the Quaternary Period.

(17) The presence of naturally occurring materials, whether identified or undiscovered, within the site, in such form that: (I) Economic extraction is currently feasible or potentially feasible during the foreseeable future; or (ii) Such materials have greater gross value or net value than the average for other areas of similar size that are representative of and located within the geologic setting.

(18) Evidence of subsurface mining for resources within the site.

(19) Evidence of drilling for any purpose within the site.

(20) Rock or groundwater conditions that would require complex engineering measures in the design and construction of the underground facility or in the sealing of boreholes and shafts.

(21) Geomechanical properties that do not permit design of underground opening that will remain stable through permanent closure.

(22) Potential for the water table to rise sufficiently so as to cause saturation of an underground facility located in the unsaturated zone.

(23) Potential for existing or future perched water bodies that may saturate portions of the underground facility or provide a faster flow path from an underground facility located in the unsaturated zone to the accessible environment.

(24) Potential for the movement of radionuclides in a gaseous state through air-filled pore spaces of an unsaturated geologic medium to the accessible environment.

60.133 Additional design criteria for the underground facility.

(a) General criteria for the underground facility. (1) The orientation, geometry, layout, and depth of the underground facility, and the design of any engineered barriers that are part of the underground facility shall contribute to the containment and isolation of radionuclides.

(h) Engineered barriers. Engineered barriers shall be designed to assist the geologic setting in meeting the performance objectives for the period following permanent closure.

IV. ACCEPTANCE CRITERIA AND GUIDANCE

General Guidance

Chapter 8 provides a summary of postclosure performance assessment as a stand-alone document. Therefore, to provide the relevant information in this chapter, some information

developed in other parts of the License Application is repeated in Chapter 8. The majority of the information presented in this chapter was based on or extracted from the Project Integrated Safety Assessment Content Guide, Appendix A to the *Management Plan for the Development of a Project Integrated Safety Assessment (PISA)* (YMP 1997).

In general, the License Application for construction authorization must provide sufficient information to enable NRC to determine that there is reasonable assurance that the types and amounts of radioactive material described in the application can be received, possessed, and disposed of in the Yucca Mountain geologic repository operations area of the design proposed without unreasonable risk to the health and safety of the public (10 CFR 60.31(a)). To support that conclusion, this chapter of the License Application must demonstrate that a rigorous total system performance assessment for the postclosure period has determined that the release of radioactive materials to the accessible environment meets the applicable standards.

The chapter must address compliance of the individual repository subsystems and the total repository system with applicable Yucca Mountain Project requirements. It must include details of the methodology and the criteria used to select possible release scenarios, the parameters and models used to represent the scenarios, the numerical and analytical tools used to perform the analyses, and the analyses and results of the total system performance assessments. The purpose of performance assessment is to demonstrate that the release of radioactive materials to the accessible environment meets the applicable standards as established by EPA.

It is expected that the performance of the overall repository system will be required to be evaluated for a dose to the average individual in a critical group for 10,000 years after repository closure. The performance assessment will evaluate the type of dose and group as dictated by the regulations existing in the License Application. Additional estimates also must be performed to assess estimated doses at longer periods. For instance, assessments will need to address peak estimated doses for selected radionuclides as specified by the EPA requirements. The release and resulting doses must be calculated for a limited suite of scenarios that must be chosen based on probability and/or consequence. Include sensitivity studies to help determine which of the parameters or processes comprising a scenario or a suite of scenarios are most important. Doses to an average individual should also be compared to other sources such as background or other doses from other activities.

more comprehensive look at scenarios and screening

The author must identify the suite of scenarios already developed for Yucca Mountain under nominal conditions, with some possible enhancements identified by site characterization or design activities.

The chapter must describe appropriate development of parameter distribution functions for the numerical models and provide the reasoning for the distributions selected. Parameter distribution functions must be used to reflect both uncertainty and variability in site and design data used in the models. Boundary and initial conditions for the analyses must be described and justified. Detailed description of conceptual models, numerical models, and analytical models used to assess repository performance must be included. Code qualification must be discussed.

The predictions of total system performance must be combined into an appropriate measure of doses, as defined by the EPA-developed governing standard. The analysis set must be

constructed from nominal and disturbed scenarios as identified previously. Include the methodology used to assess doses, including assumptions regarding pathways, location, or group. The total system performance assessment for the License Application must show some comparisons of current analyses with earlier analyses.

In the appropriate section of Chapter 8 of the License Application, identify the work that is incomplete, that is in process of completion, or that must be obtained. Identify information that needs verification. This information, including schedule of completion, will also be included in a table in Chapter 1. Remember that NRC will check for completion of such items before issuing the license to receive and possess radioactive waste, and the author must not commit to unnecessary information for final licensing or information incomplete at the time of repository startup.

Potential Modifications of Requirements

The author should be aware that NRC may accept subsystem performance that does not meet the regulatory performance objectives of 10 CFR 60.113. This allowance is stated as follows:

> **10 CFR 60.113(b)**
> On a case-by-case basis, the Commission may approve or specify some other radionuclide release rate, designed containment period or pre-waste-emplacement groundwater travel time, provided that the overall system performance objective, as it relates to anticipated processes and events, is satisfied.

Should any of the above subsystem performance values be outside of 10 CFR 60.113 limits, provide justification for the different values in the section of Chapter 8 that discusses these objectives.

Regulatory Guides

A review of the abstracts for regulatory guides found no regulatory guides pertaining to the performance of performance assessments or probabilistic risk assessments are appropriate for citing in Chapter 8 of the License Application. The author must be aware of the methods, techniques, and results of performance assessments performed under the direction of NRC and published in NUREG reports (NRC 1992, NRC 1995) and of earlier Yucca Mountain Project performance assessments, and must make comparisons of models and results whenever possible. The author should also review NUREG/CR-3332 (NRC 1983).

8.0 PURPOSE, SUMMARY, AND CONCLUSIONS

State the purpose of Chapter 8 of the License Application, which is to demonstrate that DOE has performed postclosure performance assessment of the repository and that the predicted performance will enable NRC to make a finding, with reasonable assurance, that the Monitored Geologic Repository meets regulatory requirements and performance objectives. Provide a summary of how the information in Chapter 8 relates to the ability of a repository to safely operate. Summarize how the information in Chapter 8 relates to the postclosure standards and

regulations. Discuss the organization of the chapter. Provide a summary of conclusions supporting the intent of this chapter and a description of important uncertainties and unknowns that affect the conclusions.

Identify the subissues of the NRC KTIs that are addressed in this chapter. Refer to the more detailed discussions in identified sections/subsections that follow for information regarding status of the subissues and for discussion of the acceptance criteria applicable to this chapter. M&O Licensing can provide copies of the reports for KTI information regarding how DOE has dispositioned and/or responded to each related issue resolution status report (IRSR).

NRC KTIs

Discuss in general terms how the following NRC KTIs have been addressed. Include a discussion of how NRC sees the issues as discussed in its related IRSR, and provide the DOE perspective on the status of resolution of the issue/subissue/acceptance criteria at the time of preparation of the License Application. Address each of the following acceptance criteria in a manner that demonstrates resolution of each criterion. Any aspect of any criteria not resolved should be the subject of plans to obtain additional information and should be included in the description of such activities in Section 11.11 of the technical guidance document.

The acceptance criteria text that follows each KTI below is a direct quote from the most recent relevant IRSR. The discussion provides the NRC perspective on what DOE should have provided to resolve the related subissue. (See the technical guidance document introduction for discussion regarding why some acceptance criteria from the referenced document may not be listed below.)

KTI: Container Life and Source Term

Acceptance Criteria Common to All Subissues:

(1) The collection and documentation of data, and the development and documentation of analyses, methods, models, and codes were obtained under approved quality assurance and control procedures and standards.

(2) If used, expert elicitations were conducted and documented in accordance with the guidance in NUREG-1563 (NRC 1996), or other acceptable approaches.

(4) Sensitivity and uncertainty analyses (including consideration of alternative conceptual models) are used to determine whether additional new data are needed to better define ranges of input parameters.

(5) Parameter values, assumed ranges, test data, probability distributions, and bounding assumptions used in the models are technically defensible and reasonably account for known uncertainties.

(6) Mathematical model limitations and uncertainties in modeling are defined and documented.

(7) Primary and alternative modeling approaches consistent with available data and current scientific understanding are investigated, and their results and limitations are appropriately considered in evaluating the subissue.

KTI: Total System Performance Assessment and Integration

The following acceptance criteria are programmatic and applicable to all key elements of system abstraction.

Acceptance Criteria:

Criterion Pl: Abstractions of data and process models are acceptable if they are developed and documented under acceptable quality assurance procedures.

Criterion P2: Use of formal expert elicitations to synthesize data and develop models and input for abstractions in total system performance assessment is acceptable if the elicitations are conducted and documented under acceptable procedures.

In addition to the two programmatic acceptance criteria noted above, there are five technical acceptance criteria.

Acceptance Criteria:

Criterion Tl: Data and Model Justification. Sufficient data (field, experimental, and/or natural analog data) are available to adequately support the conceptual models, assumptions, boundary conditions and define all relevant parameters implemented in the total system performance assessment.

Criterion T2: Data Uncertainty and Verification. Parameter values, assumed ranges, probability distributions, and/or bounding assumptions used in the total system performance assessment are technically defensible and reasonably account for uncertainties and variabilities.

Criterion T3: Model Uncertainty. Alternative modeling approaches consistent with available data and current scientific understanding are investigated and results and limitations appropriately considered in the abstractions.

Criterion T4: Model Verification. Models implemented in the total system performance assessment provide results consistent with output of detailed process models or empirical observations, or both.

Criterion T5: Integration. Total system performance assessment adequately incorporates important design features, physical phenomena, and couplings and uses consistent and appropriate assumptions throughout the abstraction process.

8.1 INTRODUCTION

8.1.1 Purpose

State the purpose of the performance assessment. State that the evaluation must be performed according to Yucca Mountain Project interim total system performance requirements. As noted in 10 CFR 60.134 (h), state that the engineered barrier system and the geologic setting are to work together as a system in meeting NRC postclosure performance objectives. The performance assessment must demonstrate that there is reasonable assurance that the types and amounts of radioactive material can be safely disposed of in the repository (10 CFR 60.31(a)). Address how the information presented in this chapter relates to EPA standards and requirements, and the KTIs. Present rationale and justification of all inputs and methodologies used in the total system performance assessment for the License Application.

8.1.2 Scope

Summarize the extent and limits of the performance assessment.

8.1.3 Basic Approach

Describe the components of the overall waste disposal system. Address how the information presented in this section relates to EPA standards, and the relative roles of the different components in addressing these standards. Outline the compliance demonstration logic. Address how the information presented in this section relates to NRC KTIs. Describe the approaches to development and screening of scenarios, hierarchy of models used in performance assessment, and approaches to testing alternative models. Discuss, at summary level, system and subsystem models, process models, and models of potentially disruptive features, events, and processes. Include a discussion of sensitivity analysis approaches, uncertainty analysis approaches, and the role of bounding and conservative performance assessments.

For structures, systems, and components subject to *Quality Assurance Requirements and Description* (DOE 1998), identify the verification and validation status of any computer codes discussed in this chapter and identify data that requires qualification. Identify explicitly (or by reference to another chapter) the extent of compliance of any peer review results used as input with NUREG-1297, *Peer Review for High-Level Nuclear Waste Repositories* (NRC 1988), or expert elicitation results used as input with NUREG-1563, *Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program* (NRC 1996).

8.2 SYSTEM AND SUBSYSTEM DESCRIPTIONS

need to look at overall level of conservatism

Provide a description of all the models and databases used in the performance assessments, including both the process-level models and the abstracted models. The author should coordinate section development with the authors of Chapters 3, 4, and 5. Provide the technical bases for the abstracted models, including any conservative assumptions used in developing the abstracted models, and comparison of results to corresponding process models and/or direct observations. Present and justify the basis for parameter distributions used in the abstracted models. Discuss the reliability, testing, and identification of alternative models for each process,

agree at preferred models

such as unsaturated zone flow. For each subsection in Section 8.2, explain the relevance of the information presented to overall repository performance, or NRC regulations. Provide a description of the model and code verification and validation program.

8.2.1 Site Description and Models

Describe in some detail the various site models used for performance assessment, including the geologic framework, saturated zone flow models, unsaturated zone aqueous and gaseous flow models, unsaturated zone thermo-hydrologic models, saturated zone transport models, unsaturated zone aqueous and gaseous transport models, and climate change models. Refer the reader to Chapter 3 of the License Application and call attention to these features from the site description that are addressed in the modeling process.

Demonstrate compliance with 10 CFR 60.122 (a), which states that the geological setting contains an appropriate combination of the favorable geological conditions listed in 10 CFR 60.122(b) and engineered barrier system design features so that there is a reasonable assurance that performance objectives can be met. The extent of existence of the eight types of favorable conditions is described in Chapter 3. The analytical modeling and qualitative discussions of those credited in the total system performance assessment are to be discussed here.

Compliance with 10 CFR 60.122(c) (1) through (24) must be demonstrated. This requirement lists 24 types of potentially adverse conditions. These are described and their presence is evaluated in terms of hazard in Chapter 3. Potentially adverse conditions that are present must be shown by analysis not to significantly affect the ability of the geologic repository to meet the performance objectives relating to isolation of the waste. The intent is to demonstrate that the effect of each potentially adverse condition is compensated by the presence of a combination of the favorable characteristics so that the performance objectives relating to isolation of the waste are met. The potentially adverse conditions with high probabilities of occurring or large consequences must be addressed quantitatively, and the total system performance assessment models must be presented here. Potentially adverse conditions that have sufficiently small frequencies or small consequences must be addressed qualitatively here, and favorable combinations of characteristics need not be evaluated.

Provide all information required per 10 CFR 60.21.c, including geology, favorable and adverse conditions, anticipated and unanticipated conditions, effectiveness of engineered barriers, and an explanation of the measures used to support the models.

8.2.2 Repository Description and Models

Describe the repository models used in performance assessment, including the repository layout, thermal loading designs, repository seal/backfill designs, and near-field environment models. Refer the reader to Chapter 4 of the License Application for the repository description and call attention to aspects addressed in the modeling. Include a discussion of thermo-hydrologic models.

8.2.3 Waste Package/Engineered Barrier System Description and Models

Describe the waste package/engineered barrier system and models used for performance assessment, including the waste package designs, waste package emplacement designs, waste inventory, waste package degradation models, waste form alteration/dissolution models, and waste package/engineered barrier system release models. Refer the reader to Chapter 5 of the License Application for the waste package and engineered barrier system descriptions, and call attention to aspects addressed in the modeling.

NRC KTIs

Discuss in general terms how the following NRC KTIs have been addressed. Include a discussion of how NRC sees the issue as discussed in its related IRSR, and provide the DOE perspective on the status of resolution of the issue/subissue/acceptance criteria at the time of preparation of the License Application. Address each of the following acceptance criteria in a manner that demonstrates resolution of each criterion. Any aspect of any criteria not resolved should be the subject of plans to obtain additional information and should be included in the description of such activities in Section 11.11 of the License Application.

The acceptance criteria text that follows each KTI below is a direct quote from the most recent relevant IRSR. The discussion provides the NRC perspective on what DOE should have provided to resolve the related subissue. (See the technical guidance document introduction for discussion regarding why some acceptance criteria from the referenced document may not be listed below.)

KTI: Container Life and Source Term

Subissue 1: What are the Effects of Corrosion on the Lifetime of the Containers and the Release of Radionuclides to the Near-Field Environment?

Acceptance Criteria:

This subissue relates to the adequacy of the DOE consideration of the effects of corrosion on the lifetime of the containers and the release of radionuclides from the engineered barrier system to the near-field environment.

(2) The DOE numerical corrosion models are adequate representations of expected container performance that are not likely to underestimate the actual performance of the containers in the repository environment.

(3) DOE has considered the compatibility of container materials and container fabrication processes in the performance of their intended waste isolation function. Specifically, the waste package has been designed to satisfy the appropriate sections of the rule for disposal of high-level radioactive waste at Yucca Mountain (in the current non-site specific 10 CFR 60, this is Section 60.135, "Criteria for the waste package and its components").

Subissue 2: What are the Effects of Materials Stability and Mechanical Failure on the Lifetime of the Containers and the Release of Radionuclides to the Near-Field Environment?

The acceptance criteria will be added to this document in a future revision after it is provided by revision to the related IRSR.

Subissue 3: Is Spent Nuclear Fuel Sufficiently Resistant to Contribute to the Control of Radionuclide Releases to the Near-Field Environment?

The acceptance criteria will be added to this document in a future revision after it is provided by revision to the related IRSR.

Subissue 4: Is High-Level Radioactive Waste Glass Sufficiently Resistant to Contribute to the Control of Radionuclide Releases to the Near-Field Environment?

The acceptance criteria will be added to this document in a future revision after it is provided by revision to the related IRSR.

KTI: Total System Performance Assessment and Integration

Subissue 1: Engineering Systems; Section 1: Engineered Barriers; Element 1: Waste Package Corrosion (Temperature, Humidity, and Chemistry)

Acceptance Criteria:

The DOE approach in abstracting waste package corrosion in total system performance assessment for the proposed repository at Yucca Mountain is satisfactory if the following acceptance criteria are met:

Criterion Tl: Sufficient data (field, experimental, and/or natural analog data) are available to adequately define relevant parameters and conceptual models necessary for developing the waste package corrosion abstraction in total system performance assessment.

Criterion T2: Parameter values, assumed ranges, probability distributions, and/or bounding assumptions used in the waste package corrosion abstraction, such as the critical relative humidity, material properties, pH, and chloride concentration are technically defensible and reasonably account for uncertainties and variabilities.

Criterion T3: Alternative modeling approaches consistent with available data and current scientific understanding are investigated and results and limitations appropriately factored into the waste package corrosion abstraction.

Criterion T4: Waste package corrosion abstraction output is verified through comparison to output of detailed process models and/or empirical observations (laboratory testings or natural analogs, or both).

8.2.4 Biosphere Models

Describe the biosphere models used in the performance assessment, including dose conversion models, pathways, and uptake models. Include a discussion of the surveys and other data collection. Discuss the evaluation of accumulation of radionuclides in the soil.

8.2.5 Description of Potentially Disruptive Features, Events, and Processes, and Their Corresponding Models

Provide a description of the various disruptive events and models used in performance assessment, including, as applicable, basaltic volcanism models, recurrence models, direct effect models, and indirect effect models as developed for previous total system performance assessments and subsequent work. Describe the screening of events and processes, and the development of the scenarios. Describe the human intrusion models, as required by the applicable site-specific Yucca Mountain standard and requirements. Consistency with the definitions of both anticipated and unanticipated processes and events, as defined in 10 CFR 60.2, must be maintained.

8.3 EVALUATION OF UNDISTURBED PERFORMANCE

Provide an analyses to determine the degree to which each of the favorable and potentially adverse conditions applicable to the nominal case, if present, has been characterized, and the extent to which it contributes to or detracts from isolation. For the purpose of determining the presence of the potentially adverse conditions, investigations shall extend from the surface to a depth sufficient to determine critical pathways for radionuclide migration from the underground facility to the accessible environment. Potentially adverse conditions shall be investigated outside of the postclosure controlled area if they affect isolation within the postclosure controlled area (10 CFR 60.21(c)(1)(ii)(B)).

Provide an evaluation of the performance of the geologic repository for the period after permanent closure, assuming anticipated processes and events, giving the rates and quantities of releases of radionuclides to the accessible environment as a function of time (10 CFR 60.21(c)(1)(ii)(C)). Anticipated events include climate changes.

Demonstrate compliance with 10 CFR 60.112. Demonstrate that the geologic setting was selected and the engineered barrier system and the shafts, boreholes, and seals were designed to ensure that releases of radioactive materials to the accessible environment following permanent closure conform to such generally applicable environmental standards for radioactivity as established by EPA with respect to both anticipated processes and events and unanticipated processes and events.

Demonstrate compliance with additional performance requirements promulgated by NRC under 10 CFR 60.113(c) (if any). It is not possible to predict such requirements at this time; check available online regulations to confirm if NRC has issued additional subsystem performance objectives as per this rule.

Provide an explanation of measures used to support the models used in the performance assessment. Support the analyses and models that were used to predict future conditions and changes in the geologic setting by using an appropriate combination of such methods as field tests, in situ tests, laboratory tests which are representative of field conditions, monitoring data, and natural analog studies (10 CFR 60.21(c)(1)(ii)(F)).

Provide an analysis of the potentially adverse conditions with high probabilities of occurring or large consequences; show that such conditions are addressed quantitatively in the total system performance assessment models.

8.3.1 Waste Package/Engineered Barrier System Performance

Discuss waste package degradation, including corrosion-allowance barrier degradation, corrosion-resistant barrier degradation, cladding degradation, and waste package degradation history, as applicable. Discuss radionuclide mobilization, including prompt-release inventory, alteration of the waste form, dissolution of the waste form, and solubility of radionuclides. Discuss the release from the engineered barrier system, including a discussion of advective/diffusive transport from the waste package, advective/diffusive transport from the engineered barrier system, colloidal transport from the engineered barrier system, and peak release rate from the engineered barrier system. Refer the reader to Chapter 5 of the License Application for additional information on waste package degradation.

Provide an evaluation of the effectiveness of engineered and natural barriers, including barriers that may not be a part of the geologic repository operations area, against the release of radioactive material to the environment (10 CFR 60.133 (a)(1)). The analysis must also include a comparative evaluation of alternatives to the major design features that are important to waste isolation, with particular attention to the alternatives that would improve radionuclide containment and isolation (10 CFR 60.21(c)(1)(ii)(D)). Isolation by limiting the contact with water should also be discussed.

Show compliance with 10 CFR 60.31(a)(2) and 10 CFR 60.102(e)(1), that the engineered barrier system can contain the wastes for the first several hundred years of the containment period. Demonstrate that following the containment period, the engineered barrier system works to control the release of radioactive material to the accessible environment (10 CFR 60.102(e)(2)).

Demonstrate that the waste package and engineered barrier system satisfy the requirements in 10 CFR 60.113 as follows:

- Satisfying 10 CFR 60.113(a)(1)(i)(A): Containment of high-level radioactive waste will be substantially complete during the period when radiation and thermal conditions in the engineered barrier system are dominated by fission product decay.

- Satisfying 10 CFR 60.113(a)(1)(i)(B): The engineered barrier system shall be designed so that, assuming anticipated processes and events, any release of radionuclides from the engineered barrier system shall be a gradual process which results in small fractional releases to the geologic setting over long times.

- Satisfying 10 CFR 60.113(a)(1)(ii)(A): In satisfying the preceding requirement, the engineered barrier system shall be designed, assuming anticipated processes and events, so that containment of high-level radioactive waste within the waste packages will be substantially complete for a period to be determined by NRC taking into account the factors specified in Section 60.113(b), provided that such period shall be not less than 300 years nor more than 1,000 years after permanent closure of the geologic repository.

- Satisfying 10 CFR 60.113(a)(1)(ii)(B): In satisfying the preceding requirement, the engineered barrier system shall be designed, assuming anticipated processes and events, so that the release rate of any radionuclide from the engineered barrier system following the containment period shall not exceed one part in 100,000 per year of the inventory of that radionuclide calculated to be present at 1,000 years following permanent closure, or such other fraction of the inventory as may be approved or specified by NRC; provided that this requirement does not apply to any radionuclide which is released at a rate less than 0.1 percent of the calculated total release rate limit. The calculated total release rate limit shall be taken to be one part in 100,000 per year of the inventory of radioactive waste originally emplaced in the underground facility that remains after 1,000 years of radioactive decay.

Demonstrate that the orientation, geometry, layout, depth of the underground facility, and the design of any engineered barriers that are part of the underground facility will contribute to the containment and isolation of radionuclides. Demonstrate that the engineered barrier system will assist the geologic setting in meeting the performance objectives for the period following permanent closure.

Discuss NRC KTIs (NRC 1997) dealing with the engineered barrier system performance. While these are not regulations, they represent issues that NRC will take special care in reviewing. The issues are repository design, thermal-mechanical effects, and container life and source term. For each issue that has not been discussed in previous chapters, address the acceptance criteria in the IRSR.

8.3.2 Natural Barrier System Performance

Demonstrate that the geologic setting exhibits an appropriate combination of the favorable conditions specified in 10 CFR 60.122(b)(1) through 10 CFR 60.122(b)(8) so that, together with the engineered barriers system, there is reasonable assurance that the performance objectives related to isolation are met (10 CFR 60.122(a)(1)). Demonstrate that the geologic setting is such that the isolation objectives are met, even with the presence of anticipated adverse conditions, included in 10 CFR 60.122(c)(1) through 10 CFR 60.122(c)(24) (see 10 CFR 60.122(a)(2)).

Building on, and consistent with, the information discussed in greater detail in Chapter 3, discuss the unsaturated zone gaseous flow and transport, including a discussion of advective flow, retardation, cumulative gaseous releases to the accessible environment, and peak gaseous release rates to the accessible environment. Discuss unsaturated zone aqueous flow and transport, including a discussion of advective flow, fracture-matrix interaction, matrix diffusion, dispersion, retardation, and aqueous releases to the water table. Discuss saturated zone flow and transport, including a discussion of advective flow, fracture-matrix interaction, matrix diffusion,

dispersion retardation, cumulative aqueous releases to the accessible environment, and peak aqueous release rate at the accessible environment. Show that pre-waste-emplacement groundwater travel time along the fastest path of likely radionuclide travel from the disturbed zone to the accessible environment will be at least 1,000 years (10 CFR 60.113(a)(2)). Describe the critical group in the accessible environment as required by standards and regulations and show how this group is used in performance assessment calculations. Discuss biosphere transport, including all assumptions regarding biosphere pathways, dose conversion factors, and doses to individuals or groups that must be available in a preliminary form. Include analyses of doses for the total system and any components required.

NRC KTIs

Discuss in general terms how the NRC KTIs dealing with natural barrier system performance have been addressed. Include a discussion of how NRC sees the issues as discussed in its related recent IRSR, and provide the DOE perspective on the status of resolution of the issue/subissue/acceptance criteria at the time of preparation of the License Application. Address each of the acceptance criteria in these subissues in a manner that demonstrates resolution of each criterion. Any aspect of any criteria not resolved should be the subject of plans to obtain additional information and should be included in the description of such activities in Section 11.11 of the License Application.

The acceptance criteria text that follows each KTI below is a direct quote from the most recent IRSR on the KTIs. The discussion provides the NRC perspective on what DOE should have provided to resolve the related subissue. (See the technical guidance document introduction for discussion regarding why some acceptance criteria from the referenced document may not be listed below.)

KTI: Total System Performance Assessment and Integration

Subissue 1: Engineering Systems; Section: Engineered Barriers; Element 4: Radionuclide Release Rates and Solubility Limits

Acceptance Criteria:

The DOE approach in abstracting radionuclide release rates and solubility limits in total system performance assessment for the proposed repository at Yucca Mountain is satisfactory if the following acceptance criteria are met.

Criterion TI: Sufficient data (field, experimental, and/or natural analog data) are available to adequately define relevant parameters and conceptual models necessary for developing radionuclide release rates and solubility limits abstracted in total system performance assessment.

Criterion T2: Parameter values, assumed ranges, probability distributions and/or bounding assumptions used in the radionuclide release rates and solubility limits, abstraction, such as the pH, temperature, and amount of liquid contacting the waste

forms, are technically defensible and reasonably account for uncertainties and variabilities.

Criterion T3: Alternative waste form dissolution and radionuclide release modeling approaches consistent with available data and current scientific understanding are investigated and results and limitations appropriately factored into the radionuclide release rates and solubility limits abstraction.

Criterion T5: Important design features, physical phenomena and couplings, and consistent and appropriate assumptions are incorporated into the radionuclide release rates and solubility limits abstraction.

Subissue 2: Geosphere; Section 1: Unsaturated Zone Flow and Transport; Element 1: Spatial and Temporal Distribution of Flow

Acceptance Criteria:

The DOE approach in abstracting spatial and temporal distribution of flow in total system performance assessment for the proposed repository at Yucca Mountain is satisfactory if the following acceptance criteria are met:

Criterion Tl: Sufficient data (field, experimental, and/or natural analog data) are available to adequately define relevant parameters and conceptual models necessary for developing the spatial and temporal distribution of flow abstraction in total system performance assessment.

Criterion T2: Parameter values, assumed ranges, probability distributions, and/or bounding assumptions used in the spatial and temporal distribution of flow abstraction, such as the effects of climate change on infiltration, near surface influences (e.g., evapotranspiration and runoff) on infiltration, structural controls on the spatial distribution of deep percolation, and thermal reflux owing to repository heat load, are technically defensible and reasonably account for uncertainties and variabilities.

Criterion T3: Alternative modeling approaches consistent with available data and current scientific understanding are investigated and results and limitations appropriately factored into the spatial and temporal distribution of flow abstraction.

Criterion T4: Spatial and temporal distribution of flow abstraction output is verified through comparison to output of detailed process models, and/or empirical observations (laboratory testings or natural analogs, or both).

Criterion T5: Important design features, physical phenomena and couplings, and consistent and appropriate assumptions are incorporated into the spatial and temporal distribution of flow abstraction.

Subissue 2: Geosphere; Section 2: Saturated Zone Flow and Transport; Element 1: Flow Rates in Water Production Zones

Acceptance Criteria:

The DOE approach in abstracting flow rates in water-production zones in total system performance assessment for the proposed repository at Yucca Mountain is satisfactory if the following acceptance criteria are met:

Criterion Tl: Sufficient hydrogeologic data (field, experimental, and/or natural analog data) are available to adequately define relevant parameters and conceptual models necessary for developing the flow rates in water-production zones abstraction in total system performance assessment.

Criterion T2: Parameter values, assumed ranges, probability distributions, and/or bounding assumptions used in the flow rates in the water-production zones abstraction, such as the effect of climate change on the saturated zone fluxes and water table level and well pumping practices, are technically defensible and reasonably account for uncertainties and variabilities.

Criterion T3: Alternative modeling approaches consistent with available data and current scientific understanding are investigated and results and limitations appropriately factored into the flow rates in water-production zones.

Criterion T4: Flow rates in water-production zones abstraction output are verified through comparison to output of detailed process models, and/or empirical observations (laboratory testings or natural analogs, or both).

Criterion T5: Important site (geologic and hydraulic) features, physical phenomena and couplings, and consistent and appropriate assumptions are incorporated into the flow rates in the water-production zones abstraction.

KTI: Unsaturated and Saturated Flow Under Isothermal Conditions

Subissue 2: What are the Likely Hydrologic Effects Of Climate Change?

Acceptance Criteria:

It will be necessary for DOE to develop one or more representations of future climate, to estimate the ranges of future precipitation, temperature, and water-table rise at Yucca Mountain. Water-table rise would clearly be an effect of the climate changing to cooler and wetter conditions. Changes in precipitation and temperature will be discussed as though they were also effects of climate change, but the staff recognizes that climate itself is largely defined by prevailing conditions of precipitation and temperature. Staff will review the DOE future climate representations to determine whether they are consistent with known trends of past climatic variation. The following acceptance criteria apply:

Based on staff judgment and analysis, determine whether DOE has adequately incorporated future climate changes and associated effects in its performance assessments. Current information does not support an assumption that present-day climate will persist unchanged for 10,000 years or more. The staff should keep in mind that 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.

Subissue 3: What Is The Estimated Amount and What is the Spatial Distribution of Present-Day Shallow Groundwater Infiltration?

Acceptance Criteria:

The staff's technical review of the DOE treatment of present-day shallow infiltration will be based on an evaluation of the completeness and applicability of the data and evaluations presented by DOE. It is expected that DOE will summarize or document the results of all significant infiltration related studies that have been conducted in the Yucca Mountain vicinity.

The DOE estimates of the probability distribution or upper bound for present-day shallow infiltration need not be refined further if DOE demonstrates through total system performance assessment and associated sensitivity analyses that such refinements will not significantly alter the estimate of total system performance.

KTI: Evolution of the Near-Field Environment

Subissue 1: The Effects of Coupled Processes on the Rate of Seepage into the Repository

Acceptance Criteria:

DOE has identified seepage of water into waste emplacement drifts as a factor that is important to waste containment and isolation. Important factors affecting seepage are: groundwater flux at the depth of the repository horizon, fracture density and physical properties, presence or absence of fracture coatings, rock heterogeneity, moisture content, existence of fast pathways, fluid density gradients, and others. Several aspects of seepage, particularly the quantity and chemistry of water contacting waste forms and the spatial distribution of flow, are included as elements of the NRC performance assessment subsystem abstraction. DOE must adequately estimate the quantity and chemistry of seepage and appropriately consider seepage in its assessments of waste containment and isolation.

DOE included the following relevant processes and any others that may affect seepage in its analyses:

(i) thermal-hydrologic effects on liquid flow;

(ii) effects of fracture matrix interaction. Analyses were completed to determine effects of coupled processes on seepage.

Subissue 3: The Effects of Coupled Processes on the Rate of Release of Radionuclides from Breached Waste Packages

Acceptance Criteria:

DOE has identified radionuclide mobilization from the waste form as a key factor affecting dose. Radionuclide release rates and solubility limits constitute one of the NRC key elements of subsystem abstraction for performance assessment. DOE must adequately estimate rates of radionuclide release and appropriately consider these rates and processes affecting them in its assessments of waste containment and isolation.

DOE is to include the following processes and any others that may affect rate of release in its analyses:

(i) thermal-hydrologic effects on liquid flow;

(ii) hydrothermal-chemical effects, such as changes in groundwater chemistry that may affect rate of release;

(v) effects of cementitious materials on chemical conditions and hydraulic properties affecting rate of release.

Data and models have been collected, developed, and documented under acceptable quality assurance procedures.

NRC staff should verify that there are no deficiency reports concerning data quality related to effects of coupled processes on the rate of radionuclide release that have not been closed.

If used, expert elicitations were conducted and documented in accordance with the guidance in NUREG-1563 (NRC 1996), or other acceptable approaches.

Subissue 4: The Effects of Coupled Processes on Radionuclide Transport Through Engineered and Natural Barriers

Acceptance Criteria:

DOE considers radionuclide transport a key performance attribute of the proposed repository. Retardation of radionuclides in fractures in the unsaturated zone and in the saturated zone constitute two of the NRC key elements of subsystem abstraction for performance assessment. DOE must adequately estimate the radionuclide transport characteristics of the near field and appropriately consider radionuclide transport in its assessments of waste containment and isolation.

Data and models have been collected, developed, and documented under acceptable quality assurance procedures.

NRC staff should verify that there are no deficiency reports concerning data quality in relation to radionuclide transport that have not been closed.

If used, expert elicitations were conducted and documented in accordance with the guidance in NUREG-1563 (NRC 1996), or other acceptable approaches.

KTI: Thermal Effects on Flow

Subissue 3: Does the U.S. Department of Energy Total System Performance Assessment Adequately Account for Thermal Effects on Flow?

Acceptance Criteria:

This subissue relates to DOE demonstration of the adequacy of its total system performance assessment with respect to thermal effects on flow. However, it should be noted that resolution of the subissue is not intended to be interpreted as, or deemed to be a determination of, the acceptability of the entire DOE total system performance assessment.

Those aspects of the DOE analysis of total system performance that relate to thermal effects on flow are acceptable, if the following acceptance criteria are met:

Programmatic Elements:

The DOE analyses were developed and documented under acceptable quality assurance procedures.

Expert elicitations may be used for, but not necessarily limited to, justifying the use of abstracted models in the DOE total system performance assessment. All expert elicitations are conducted and documented in accordance with the guidance in NUREG-1563 (NRC 1996), or other acceptable procedures.

8.3.3 Sensitivity of Undisturbed Performance Results

Summarize the sensitivity studies, including sensitivity of results to uncertain near-field environments, sensitivity of results to uncertain waste package/engineered barrier system performance, sensitivity of results to uncertain gaseous flow and transport, sensitivity of results to uncertain unsaturated zone flow and transport, sensitivity of results to uncertain saturated zone flow and transport, sensitivity of results to uncertain biosphere transport, sensitivity of results to an alternative repository thermal loads, and sensitivity to climate changes.

The potentially adverse conditions with high probabilities of occurring or large consequences must be addressed quantitatively. Present the total system performance assessment results here.

8.4 EVALUATION OF POTENTIALLY DISTURBED PERFORMANCE

Provide an evaluation of the performance of the geologic repository for the period after permanent closure, assuming unanticipated processes and events, giving the rates and quantities of releases of radionuclides to the accessible environment as a function of time (10 CFR 60.21(c)(1)(ii)(C)). Unanticipated processes and events include seismic events (seismic hazard to be defined by author), igneous activity, and human intrusion.

Provide an evaluation of the effectiveness of engineered and natural barriers, including barriers that may not be a part of the geologic repository operations area, against the release of radioactive material to the environment from unanticipated events or processes (10 CFR 60.21(c)(1)(ii)(C)).

Explain measures used to support the performance assessment models. State that analyses and models that were used to predict future conditions and changes in the geologic setting are supported by appropriate combination of such methods as field tests, in situ tests, laboratory tests which are representative of field conditions, monitoring data, and natural analog studies (10 CFR 60.21(c)(1)(ii)(F)). Provide analysis to support this position.

Demonstrate that the effect of the potentially adverse human activity or natural condition on the site has been adequately evaluated using analyses which are sensitive to the potentially adverse human activity or natural condition and assumptions which are not likely to underestimate its effect; and (a) the potentially adverse human activity or natural condition is shown by analysis not to affect significantly the ability of the geologic repository to meet the performance objectives relating to isolation of the waste, or (b) the effect of the potentially adverse human activity or natural condition is compensated by the presence of a combination of the favorable characteristics so that the performance objectives relating to isolation of the waste are met (10 CFR 60.122(a)(2), Siting Criteria).

8.4.1 Identification of Features, Events, and Processes

Identify significant features, events, and processes that form scenarios associated with potential site disturbances. Establish initial screening criteria, probabilities, and consequences for each type of disturbance. Explain the relevance of the information presented to overall repository performance (10 CFR 60.21(c)(ii)(C)).

Identify the suite of scenarios currently developed for Yucca Mountain under nominal conditions, with some possible enhancements identified by site characterization or design activities. This should include an initial plan for identifying specific criteria to be used for screening nominal case scenarios to eliminate those that are logically or physically unrealistic, or are expected to result in trivial consequences. Document the rationale for the elimination or retention of any scenario. The final suite of scenarios used as a basis for calculations must be identified in detail. Modify scenarios related to disturbances, such as seismic events or volcanism, from the previously existing suite only if site characterization presents major changes in interpretations of the probability of occurrence or the magnitude of an event.

Address probabilities and uncertainties associated with the scenarios and associated parameters to be analyzed. Discuss initial work on the techniques used to estimate probabilities such as predictive modeling and expert judgment, in the context of 10 CFR 60.101, and 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 exists. For all cases analyzed in the total system performance assessment for the License Application, identify the factual bases and rationale for the accepted values.

8.4.2 Screening of Features, Events, and Processes

Summarize the screening criteria, features, events, and processes selected, and basis for eliminating features, events, and processes. Discuss the combination of features, events, and processes into scenarios.

8.4.3 Evaluation of Disturbed Scenario Probabilities

Summarize the event probabilities for the basaltic volcanism scenarios, tectonic scenarios, and human interference scenarios. The consequence of these scenarios must be discussed. Demonstrate the sensitivity of disturbed performance results to key assumptions.

Demonstrate that after permanent closure, generally applicable environmental standards for radioactivity as may have been established by EPA with respect to unanticipated processes and events can be met (10 CFR 60.112).

Demonstrate, assuming anticipated processes and events, that containment of high-level radioactive waste within the waste packages will be substantially complete for a period to be determined by NRC, provided that such period shall be not less than 300 years nor more than 1,000 years after permanent closure of the geologic repository (10 CFR 60.113(a)(1)(ii)(A)).

Demonstrate, assuming anticipated processes and events, that the release rate of any radionuclide from the engineered barrier system following the containment period shall not exceed one part in 100,000 per year of the inventory of that radionuclide calculated to be present at 1,000 years following permanent closure, or such other fraction of the inventory as may be approved or specified by NRC; provided that this requirement does not apply to any radionuclide which is released at a rate less than 0.1 percent of the calculated total release rate limit. The calculated total release rate limit shall be taken to be one part in 100,000 per year of the inventory of radioactive waste, originally emplaced in the underground facility, that remains after 1,000 years of radioactive decay (10 CFR 60.113(a)(1)(ii)(B)).

NRC KTIs

Discuss in general terms how the following NRC KTI has been addressed. Include a discussion of how NRC sees the issue as discussed in its related IRSR, and provide the DOE perspective on the status of resolution of the issue/subissue/acceptance criteria at the time of preparation of the License Application. Address each of the following acceptance criteria in a manner that demonstrates resolution of each criterion. Any aspect of any criteria not resolved should be the

subject of plans to obtain additional information and should be included in the description of such activities in Section 11.11 of the License Application.

The acceptance criteria text that follows each KTI below is a direct quote from the most recent relevant IRSR. The discussion provides the NRC perspective on what DOE should have provided to resolve the related subissue. (See the technical guidance document introduction for discussion regarding why some acceptance criteria from the referenced document may not be listed below.)

KTI: Igneous Activity

Subissue 1: Probability

Acceptance Criteria:

Estimates of the probability of future igneous activity in the Yucca Mountain region will be acceptable, provided that:

(6) The probability values used by DOE in performance assessments reflect the uncertainty in the DOE probabilistic volcanic hazard estimates.

(7) The values used (single values, distributions, or bounds on probabilities) are technically justified and account for uncertainties in probability estimates.

Subissue 2: Consequences

The acceptance criteria will be added to this document in a future revision after it is provided by revision to the related IRSR.

KTI: Repository Design and Thermal Mechanical Effects

Subissue 2: Design of the Geologic Repository Operations Area for the Effects of Seismic Events and Direct Fault Disruption

Acceptance Criteria:

The staff will find the methodology proposed in the topical report on design methodology adequate for use in Exploratory Studies Facility and repository design if the following criteria are satisfied:

(8) Any major assumptions or limitations to the proposed methodology are identified and the implications regarding design and performance are discussed in the topical report.

(10) The contents of the topical report on design methodology are consistent with the contents of the topical report on the proposed DOE methodology to assess seismic hazards. Also, taken together, the two topical reports support the development of inputs in accordance with the third topical report that deals with vibratory ground

motion and fault displacement inputs to be used in repository design and performance assessments.

KTI: Total System Performance Assessment and Integration

Subissue 1: Engineering System; Section 1: Engineered Barriers; Element 2: Mechanical Disruption of Waste Packages (Seismicity, Faulting, Rockfall, and Dike Intrusion)

Acceptance Criteria:

The DOE approach to abstracting mechanical disruption of waste packages in total system performance assessment for the proposed repository at Yucca Mountain is satisfactory if the following acceptance criteria are met:

Criterion Tl: Sufficient data (field, experimental, and/or natural analog data) are available to adequately define relevant parameters and conceptual models necessary for developing mechanical disruption of waste packages' abstraction in total system performance assessment.

Criterion T2: Parameter values, assumed ranges, probability distributions, and/or bounding assumptions used in the mechanical disruption of waste packages' abstraction, such as probabilistic seismic hazard curves, probability of dike intrusion, and the probability and amount of fault displacement, are technically defensible and reasonably account for uncertainties and variabilities.

Criterion T4: Mechanical disruption of waste packages' abstraction output is verified through comparison to output of detailed process models, and/or empirical observations (laboratory testings or natural analogs, or both).

Subissue 1: Engineering System; Section 1: Engineered Barriers; Element 3: Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms

Acceptance Criteria:

The DOE approach to abstracting quantity and chemistry of water contacting waste packages and waste forms in total system performance assessment for the proposed repository at Yucca Mountain is satisfactory if the following acceptance criteria are met:

Criterion Tl: Sufficient data (field, experimental, and/or natural analog data) are available to adequately define relevant parameters and conceptual models necessary for developing the quantity and chemistry of water contacting waste packages and waste forms abstraction in total system performance.

Criterion T2: Parameter values, assumed ranges, probability distributions, and/or bounding assumptions used in the quantity and chemistry of water contacting waste packages and waste forms abstraction, such as the pH, chloride concentration, and amount of water flowing in and out of the breached waste package, are technically defensible and reasonably account for uncertainties and variabilities.

Criterion T3: Alternative modeling approaches consistent with available data and current scientific understanding are investigated and results and limitations appropriately factored into the quantity and chemistry of water contacting waste packages and waste forms abstraction.

Criterion T4: Output of quantity and chemistry of water contacting waste packages and waste forms abstraction are verified through comparison to output of detailed process models and/or empirical observations (laboratory testings or natural analogs, or both).

Criterion T5: Important design features, physical phenomena and couplings, and consistent and appropriate assumptions are incorporated into the quantity and chemistry of water contacting waste packages and waste forms abstraction.

Subissue 2: Geosphere; Section 3: Direct Release and Transport; Element 1: Volcanic Disruption of Waste Packages

Acceptance Criteria:

The DOE approach in abstracting the volcanic disruption of waste packages in total system performance assessment for the proposed repository at Yucca Mountain is satisfactory if the following acceptance criteria are met:

Criterion Tl: Sufficient data (field data or natural analog data, or both) are available to adequately define relevant parameters and conceptual models necessary for abstracting the volcanic disruption of waste packages in total system performance assessment.

Criterion T2: Parameter values, assumed ranges, probability distributions, and/or bounding assumptions used in abstracting the volcanic disruption of waste packages, such as the probability of volcanism, number of waste packages affected, and the amount of spent fuel particles incorporated into the tephra and ejected, are technically defensible and reasonably account for uncertainties and variabilities.

Criterion T3: Alternative modeling approaches consistent with available data and current scientific understanding are investigated and results and limitations appropriately factored into the volcanic disruption of waste packages' abstraction.

Criterion T4: Volcanic disruption of waste packages' abstraction outputs are verified through comparison to output of detailed process models and/or empirical observations (laboratory testings or natural analogs, or both).

Criterion T5: Important site and design features, physical phenomena and couplings, and consistent and appropriate assumptions are incorporated into the volcanic disruption of waste packages' abstraction.

Subissue 2: Geosphere; Section 3: Direct Release and Transport; Element 2: Airborne Transport of Radionuclides

Acceptance Criteria:

The DOE approach in abstracting the airborne transport of radionuclides in total system performance assessment for the proposed repository at Yucca Mountain is satisfactory if the following acceptance criteria are met:

Criterion Tl: Sufficient data (field, experimental, and/or natural analog data) are available to adequately define relevant parameters and conceptual models necessary for developing the airborne transport of radionuclides abstraction in total system performance assessment.

Criterion T2: Parameter values, assumed ranges, probability distributions, and/or bounding assumptions used in the airborne transport of radionuclides abstraction, such as the magnitude of eruption and deposition velocity, are technically defensible and reasonably account for uncertainties and variabilities.

Criterion T3: Alternative modeling approaches consistent with available data and current scientific understanding are investigated and results and limitations appropriately factored into the airborne transport of radionuclides abstraction.

Criterion T4: Airborne transport of radionuclides abstraction output is verified through comparison to output of detailed process model or empirical observations (i.e., natural analogs), or both.

Criterion T5: Important site features, physical phenomena and couplings, and consistent and appropriate assumptions are incorporated into the airborne transport of radionuclides abstraction.

8.5 CONCLUSIONS

An executive summary of the conclusions is given in Section 8.0. Provide a more detailed discussion of these conclusions in this section, based on the information in Chapter 8, and relate these conclusions to the uncertainties in and sensitivities of the major data sources and modeling assumptions. Discuss sensitivity to major conceptual model uncertainties, major design features, and major potentially disturbed performance scenario alternatives to set previous uncertainty and sensitivity discussions into a larger context.

Summarize the findings about each NRC KTI in the conclusions section, including referencing the section of the License Application in which the subject is discussed in detail. NRC has identified 10 KTIs; while these are not regulations or requirements, they represent issues that NRC will take special care in reviewing. Authors must take special care in addressing these concerns. The KTIs (NRC 1996) are:

- Unsaturated and saturated flow under isothermal conditions
- Thermal effects on flow
- Evolution of the near-field environment
- Repository design and thermal-mechanical effects
- Container life and source term
- Radionuclide transport
- Igneous activity
- Structural deformation and seismicity
- Activities related to development of the EPA standard
- Total system performance assessment and integration

V. REFERENCES

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

40 CFR 191. Protection of Environment: Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes.

DOE (U.S. Department of Energy) 1998. *Quality Assurance Requirements and Description.* DOE/RW-0333P. Rev 8. Washington D.C.: U.S. Government Printing Office. MOL 19980601.0022.

EPAct (Energy Policy Act) 1992. Energy Policy Act of 1992. Public Law 102-486. Washington, D.C.: U.S. Government Printing Office. 233191.

NAS (National Academy of Sciences) 1995. *Technical Basis for Yucca Mountain Standards.* National Research Council for National Academy of Science. Washington, D.C.: National Academy Press for the National Academy of Sciences. 104273.

NRC (U.S. Nuclear Regulatory Commission) 1983. *Radiological Assessment, 'A Textbook of Environmental Dose Assessment.'* NUREG/CR-3332. Washington, D.C.: U.S. Government Printing Office. 223809.

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CHAPTER 9. RADIOACTIVE WASTE MANAGEMENT

I. OBJECTIVE

This chapter provides guidance for the author(s) of Chapter 9 of the License Application. The License Application will be submitted to the U.S. Nuclear Regulatory Commission (NRC) for authorization to construct a geologic high-level radioactive waste repository at Yucca Mountain.

The author(s) will be required to demonstrate that U.S. Department of Energy (DOE) has designs for radioactive waste management systems and design analyses sufficient to provide reasonable assurance of the adequacy of the design in supporting repository performance objectives and meeting regulatory design requirements. Many of the components associated with waste management systems have substantial licensing precedent, and thus the level of detail required will be moderate. Where the design of components involves considerations unique to the geologic repository operations area, the required level of detail of information to be presented and referenced in this chapter will be substantial, as discussed in more detail in the information that follows.

II. GUIDANCE TOPICS

The following topics will be discussed in Chapter 9 of the License Application:

- 9.0 PURPOSE, SUMMARY, AND CONCLUSIONS
- 9.1 SOURCE TERMS
  - 9.1.1 Liquid Low-Level Radioactive Waste Source Terms
  - 9.1.2 Solid Low-Level Radioactive Waste Source Terms
  - 9.1.3 Gaseous Waste Source Terms
  - 9.1.4 Mixed Low-Level Radioactive Waste Source Terms
- 9.2 LIQUID LOW-LEVEL RADIOACTIVE WASTE MANAGEMENT SYSTEM
  - 9.2.1 Design Bases
  - 9.2.2 System Description
  - 9.2.3 Design Evaluation
- 9.3 SOLID LOW-LEVEL WASTE MANAGEMENT SYSTEM
  - 9.3.1 Design Bases
  - 9.3.2 System Description
  - 9.3.3 Design Evaluation
- 9.4 GASEOUS WASTE MANAGEMENT SYSTEM
  - 9.4.1 Design Bases
  - 9.4.2 System Description
  - 9.4.3 Design Evaluation
- 9.5 MIXED WASTE MANAGEMENT SYSTEM
  - 9.5.1 Design Bases
  - 9.5.2 System Description
  - 9.5.3 Design Evaluation

9.6 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS
9.6.1 Design Bases
9.6.2 System Description
9.6.3 Effluent Monitoring and Sampling Operations
9.6.4 Design Evaluation

III. REGULATORY REQUIREMENTS

***** IMPORTANT NOTE *****

At the time of development of this document, it appears likely that NRC will issue a new, site-specific regulation for Yucca Mountain in 1999. At the time the new 10 CFR 63 is issued, 10 CFR 60 will be modified to state that it is not applicable to Yucca Mountain. Details about the new regulation are not known. However, it has been determined prudent to develop the working draft License Application in accordance with the best available estimate of the form of the new regulation. To avoid confusing references and to allow production of the technical guidance document in time to support training for the working draft License Application, the text of the technical guidance document that follows this note has not been revised to reflect the postulated form of the new regulation. Instead, this note provides guidance needed by working draft License Application authors to comply with management direction on addressing the new regulation. The paragraphs that follow explain changes in the regulation with respect to the existing 10 CFR 60 that affect this technical guidance document chapter and provide guidance on how to address the changes in the working draft License Application. Where guidance in this note conflicts with guidance in the remainder of this chapter of the technical guidance document, the guidance in the note is to govern. Any questions should be addressed to Management and Operating Contractor (M&O) Licensing.

Changes to Regulations Applicable to Chapter 9

The following 10 CFR 60 requirements applicable to technical guidance document Chapter 9 and listed below in Section III are assumed to not be carried over to the new regulation: 60.130, 60.131, and 60.136.

In place of these specific requirements, the new regulation is expected to require a discussion of the contribution of the individual barriers to the overall performance of the repository. The manner in which that contribution is to be described has not been specified, though an NRC review plan on the subject may be issued sometime in the future.

Guidance to Working Draft License Application Authors for Addressing Changes

- Disregard guidance in Section IV regarding demonstrating compliance with any of the regulations assumed to be deleted as described above. Instead, discuss the performance

of the engineered barrier system and its components in supporting overall repository performance.

- Address the technical issue of each of the regulations in Section III, including those expected to be deleted, but do not relate the discussion to demonstrating compliance with any of the deleted regulations. Continue to explicitly address the regulations in Section III that are not expected to be deleted.

- Detailed discussion of design criteria, design bases, and performance is still required as per the existing guidance in Section IV. Discussion should still include detailed discussions of the phenomena that affect that performance.

- No sections or subsections of Chapter 9 are expected to be deleted as a result of this note.

***** END OF NOTE *****

Sections of the regulations that are to be addressed fully or in part in Chapter 9 of the License Application are reproduced below. These are verbatim extracts from the regulations. Paragraph numbers refer to Title 10 of the U.S. Code of Federal Regulations (CFR). In addition, Appendix A provides a cross-reference between the regulations in 10 CFR 60 and the specific License Applications sections that address these regulations. Since revisions of regulations are not uncommon, verify that current versions are used.

10 CFR Part 20, Standards For Protection Against Radiation
Subpart B - Radiation Protection Programs
20.1101 Radiation protection programs.

(b) The licensee shall use, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA)...

(d) To implement the ALARA requirements of §20.1101(b), and notwithstanding the requirements in §20.1301 of this part, a constraint on air emissions of radioactive material to the environment, excluding Radon-222 and its daughters, shall be established by licensees other than those subject to §50.34a, such that the individual member of the public likely to receive the highest dose will not be expected to receive a total effective dose equivalent in excess of 10 mrem (0.1 mSv) per year from these emissions. If a licensee subject to this requirement exceeds this dose constraint, the licensee shall report the exceedance as provided in §20.2203 and promptly take appropriate corrective action to ensure against recurrence.

Subpart D - Radiation Dose Limits for Individual Member of the Public

20.1301 Dose limits for individual members of the public.

(a) Each licensee shall conduct operations so that-

(1) The total effective dose equivalent to individual members of the public from the licensed operation does not exceed 0.1 rem (1 millisievert) in a year, exclusive of the dose contributions from background radiation, any medical administration the individual has received, voluntary participation in medical research programs, and the licensee's disposal of radioactive material into sanitary sewerage in accordance with §20.2003.

(2) The dose in any unrestricted area from external sources does not exceed 0.002 rem (0.02 mSv) in any one hour.

20.1302 Compliance with dose limits for individual members of the public.

(a) The licensee shall make or cause to be made, as appropriate, surveys of radiation levels in unrestricted and controlled areas and radioactive materials in effluents released to unrestricted and controlled areas to demonstrate compliance with the dose limits for individual members of the public in §20.1301.

(b) A licensee shall show compliance with the annual dose limit in §20.1301 by-

(1) Demonstrating by measurement or calculation that the total effective dose equivalent to the individual likely to receive the highest dose from the licensed operation does not exceed the annual dose limit; or

(2) Demonstrating that-

(i) The annual average concentrations of radioactive material released in gaseous and liquid effluents at the boundary of the unrestricted area do not exceed the values specified in table 2 of Appendix B to Part 20; and

(ii) If an individual were continuously present in an unrestricted area, the dose from external sources would not exceed 0.002 rem (0.02 mSv) in an hour and 0.05 rem (0.5 mSv) in a year.

(c) Upon approval from the Commission, the licensee may adjust the effluent concentration values in Appendix B to Part 20, table 2, for members of the public, to take into account the actual physical and chemical characteristics of the effluents (e.g., aerosol size distribution, solubility, density, radioactive decay equilibrium, chemical form).

Subpart F - Surveys and Monitoring

20.1501 General.

(b) The licensee shall ensure that instruments and equipment used for quantitative radiation measurements (e.g., dose rate and

effluent monitoring) are calibrated periodically for the radiation measured.

10 CFR Part 60, Disposal of High-Level Radioactive Wastes in Geologic Repositories
Subpart B - Licenses
License Application
60.21 Content of Application...

(c) The Safety Analysis Report shall include:...

(1) A description and assessment of the site at which the geologic repository operations area is to be located with appropriate attention to those features of the site that might affect geologic repository operations area design and performance...

(ii) The assessment shall contain:...

(E) An analysis of the performance of the major design structures, systems, and components, both surface and subsurface, to identify those that are important to safety. For the purpose of this analysis, it shall be assumed that operations at the geologic repository operations area will be carried out at the maximum capacity and rate of receipt of radioactive waste stated in the application...

(2) A description and discussion of the design, both surface and subsurface, of the geologic repository operations area including: (i) the principal design criteria and their relationship to any general performance objectives promulgated by the Commission (ii) the design bases and the relation of the design bases to the principal design criteria, (iii) information relative to materials of construction (including geologic media, general arrangement, and approximate dimensions), and (iv) codes and standards that the DOE proposes to apply to the design and construction of the geologic repository operations area...

(3) A description...of the design and performance requirements for structures, systems, and components of the geologic repository that are important to safety...

(7) A description of the program for control and monitoring of radioactive effluents and occupational radiation exposures to maintain such effluents and exposures in accordance with the requirements of part 20 of this chapter...

(11) A description of design considerations that are intended to facilitate...decontamination or dismantlement of surface facilities.

Construction Authorization
60.31 Construction authorization.

Upon review and consideration of an application and environmental impact statement submitted under this part, the Commission may authorize construction if it determines:

(a) *Safety.* That there is reasonable assurance that the types and amounts of radioactive materials described in the application can be received, possessed, and disposed of in a geologic repository operations area of the design proposed without unreasonable risk to the health and safety of the public. In arriving at this determination, the Commission shall consider whether:

(1) the DOE has described the proposed geologic repository including but not limited to:... (iii) the principal architectural and engineering criteria for the design of the geologic repository operations area; ...and (v) features or components incorporated in the design for the protection of the health and safety of the public.

(2) The site and design comply with the performance objectives and criteria contained in Subpart E of this part.

Subpart E - Technical Criteria
Performance Objectives
60.111 Performance of the geologic repository operations area through permanent closure.

(a) *Protection against radiation exposures and releases of radioactive material.* The geologic repository operations area shall be designed so that until permanent closure has been completed, radiation exposures and radiation levels, and releases of radioactive materials to unrestricted areas, will be maintained within the limits specified in part 20 of this chapter and such generally applicable environmental standards for radioactivity as may have been established by Environmental Protection Agency.

Design Criteria for the Geologic Repository Operations Area
60.130 General considerations.

Pursuant to the provisions of §60.21(c)(2)(i), an application to receive, possess, store, and dispose of high-level radioactive waste in the geologic repository operations area must include the principal design criteria for a proposed facility. The principal design criteria establish the necessary design, fabrication, construction, testing, maintenance, and performance requirements for structures, systems, and components important to safety and/or important to waste isolation. Sections 60.131 through 60.134 specify minimum requirements for the principal design criteria for the geologic repository operations area.

These design criteria are not intended to be exhaustive. However, omissions in §60.131 through 60.134 do not relieve the DOE from any obligation to provide such features in a specific facility needed to achieve the performance objectives.

60.131 General design criteria for the geologic repository operations area.

(a) *Radiological protection.* The geologic repository operations area shall be designed to maintain radiation doses, levels, and concentrations of radioactive material in air in restricted areas within the limits specified in part 20 of this chapter. Design shall include:

(1) Means to limit concentrations of radioactive material in air;

(2) Means to limit the time required to perform work in the vicinity of radioactive materials, including, as appropriate, designing equipment for ease of repair and replacement and providing adequate space for ease of operation;

(3) Suitable shielding;

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

(6) A radiation alarm system to warn of significant increases in radiation levels, concentrations of radioactive material in air, and of increased radioactivity released in effluents. The alarm system shall be designed with provisions for calibration and for testing its operability.

(b) *Protection against design basis events.* The structures, systems, and components important to safety shall be designed so that they will perform their necessary safety functions, assuming occurrence of design basis events.

(c) *Protection against dynamic effects of equipment failure and similar events.* The structures, systems, and components important to safety shall be designed to withstand dynamic effects such as missile impacts, that could result from equipment failure, and similar events and conditions that could lead to loss of their safety functions.

(d) *Protection against fires and explosions.* (1) The structures, systems, and components important to safety shall be designed to perform their safety functions during and after credible fires or explosions in the geologic repository operations area.

(2) To the extent practicable, the geologic repository operations area shall be designed to incorporate the use of noncombustible and heat resistant materials...

(4) The geologic repository operations area shall be designed to include means to protect structures, systems, and components important to safety against the adverse effects of either the operation or failure of the fire suppression systems.

(e) *Emergency capability.* (1) The structures, systems, and components important to safety shall be designed to maintain control of radioactive waste and radioactive effluents, and permit prompt termination of operations and evacuation of personnel during an emergency.

(f) *Utility services.* (1) Each utility service system that is important to safety shall be designed so that essential safety functions can be performed, assuming occurrence of the design basis events.

(3) 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.

(g) *Inspection, testing, and maintenance.* The structures, systems, and components important to safety shall be designed to permit periodic inspection, testing, and maintenance, as necessary, to ensure their continued functioning and readiness.

(h) *Criticality control.* All systems for processing, transporting, handling, storage, retrieval, emplacement, and isolation of radioactive waste shall be designed to ensure that nuclear criticality is not possible unless at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. Each system must be designed for criticality safety assuming occurrence of design basis events. The calculated effective multiplication factor (k_{eff}) must be sufficiently below unity to show at least a 5 percent margin, after allowance for the bias in the method of calculation and the uncertainty in the experiments used to validate the method of calculation.

(i) *Instrumentation and control systems.* The design shall include provisions for instrumentation and control systems to monitor and control the behavior of systems important to safety, assuming occurrence of design basis events.

60.132 Additional design criteria for surface facilities in the geologic repository operations area.

(c) *Radiation control and monitoring*--(1) *Effluent control.* The surface facilities shall be designed to control the release of radioactive materials in effluents during Category 1 design basis events so as to meet the performance objectives of §60.111(a).

(2) *Effluent monitoring.* The effluent monitoring systems shall be designed to measure the amount and concentration of radionuclides in any effluent with sufficient precision to determine whether releases conform to the design requirement for effluent control. The monitoring systems shall be designed to include alarms that can be periodically tested.

(d) *Waste treatment.* Radioactive waste treatment facilities shall be designed to process any radioactive wastes generated at the geologic repository operations area into a form suitable to permit safe disposal at the geologic repository operations area or to

permit safe transportation and conversion to a form suitable for disposal at an alternative site in accordance with any regulations that are applicable.

(e) *Consideration of decommissioning.* The surface facility shall be designed to facilitate decontamination or dismantlement to the same extent as would be required, under other parts of this chapter, with respect to equivalent activities licensed thereunder.

Preclosure Controlled Area

60.136 Preclosure controlled area.

(b) The geologic repository operations area shall be designed so that, for Category 2 design basis events, no individual located on or beyond any point on the boundary of the preclosure controlled area will receive the more limiting of a total effective dose equivalent of 0.05 Sv (5 rem), or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.5 Sv (50 rem). The eye dose equivalent shall not exceed 0.15 Sv (15 rem), and the shallow dose equivalent to skin shall not exceed 0.5 Sv (50 rem). The minimum distance from the surface facilities in the geologic repository operations area to the boundary of the preclosure controlled area must be at least 100 meters.

IV. ACCEPTANCE CRITERIA AND GUIDANCE

General Guidance

This section contains the guidance for preparation of the License Application for construction authorization.

In general, the License Application for construction authorization must provide sufficient information that NRC can make a determination that there is reasonable assurance that the types and amounts of radioactive materials described in the application can be received, possessed, and disposed of in the Yucca Mountain geologic repository operations area of the design proposed without unreasonable risk to the health and safety of the public. To support that conclusion, this chapter must provide the following information:

- Identification of the source terms of the radioactive waste management systems[1]
- Description of the design bases and design of the radioactive waste management systems

[1] These source terms will form the basis for calculation of dose to the public for radioactive releases (Chapter 7) and will form part of the basis for calculation of occupational dose (Chapters 7 and 10). Chapter 9 will contain no dose estimates/assessments.

- Demonstration that the applicable design and other requirements of 10 CFR 60 have been addressed

- Description of the system for radioactive process and effluent monitoring and sampling

In addition to demonstrating compliance with the design criteria of 10 CFR 60, the License Application will demonstrate that the waste package and repository designs are consistent with the precedents set by 10 CFR 50 general design criteria to the extent the criteria are applicable. As 10 CFR 50 regulations, these criteria are not regulatory requirements for the repository, although some have 10 CFR 60 counterparts that are requirements. Section 2.1 of the License Application discusses the applicability of these criteria to the repository and waste package. Criteria applicable to the radioactive waste management systems are shown in the individual sections.

Although the amount of detail required in Chapter 9 of the License Application varies, in all cases sufficient information should be included to allow NRC to make the safety determination addressed in 10 CFR 60.31(a) (related to Sections (1)(iii), (1)(v), and (2)) and 60.101(b) to authorize construction. The information required is:

- Principal architectural and engineering criteria for the design of the geologic repository operations area

- Features or components incorporated in the design for the protection of the health and safety of the public

- Evidence that the design complies with the performance objectives and criteria contained in 10 CFR 60, Subpart E

Source term information must be detailed enough to support dose assessments performed in Chapters 7 and 10. Where design of waste systems is similar to designs elsewhere within the nuclear industry, only moderate detail is required in the description. Where the geologic repository operations area presents particularly unique challenges or application to system design, a higher level of detail is required. Identification of committed guides and standards is important. Where the License Application commits to NRC-accepted industry standards, little descriptive detail is needed. Where alternative approaches are to be used, much more description is required. It is important that specific applications of as low as reasonably achievable (ALARA) concepts to the design of the radioactive waste management systems be identified in the text to demonstrate an effective ALARA program during the design process.

Most of the guidance in this chapter, other than that specified in the CFR, is based on or extracted from the following documents:

- NUREG-1323, *License Application Review Plan for a Geologic Repository for Spent Nuclear Fuel and High-Level Radioactive Waste* (NRC 1995)

- YMP/97-01Q, *Management Plan for the Development of a Project Integrated Safety Assessment (PISA)* (YMP 1997)
- NUREG-1567, *Standard Review Plan for Spent Fuel Dry Storage Facilities* (NRC 1996b)
- Regulatory Guide DG-3003, *Format and Content for the License Application for the High-Level Waste Repository* (NRC 1990)
- Regulatory Guide 3.61, *Standard Format and Content for a Topical Safety Report for a Spent Fuel Dry Storage Cask* (NRC 1989)
- NUREG-1536, *Standard Review Plan for Dry Cask Storage Systems* (NRC 1997)

NUREG-1323 (NRC 1995) is largely incomplete and will become inapplicable to Yucca Mountain upon the expected issuance of a new 10 CFR 63 as discussed in Section III above. Information from NUREG-1323 (NRC 1995) considered likely to be applicable under the new regulation is included in the technical guidance document, though the NUREG is not referenced as an authoritative source for such information. YMP/97-01Q (YMP 1997) contains the Project Integrated Safety Assessment Content Guide that has been developed by the Yucca Mountain Project based primarily on the NRC Regulatory Guide DG-3003, *Format and Content for the License Application for the High-Level Waste Repository* (NRC 1990). The Project Integrated Safety Assessment Content Guide is the starting point for the development of the *Technical Guidance Document for License Application Preparation*. As is the case for NUREG-1323 (NRC 1995), DG-3003 (NRC 1990) will become inapplicable to Yucca Mountain upon the expected issuance of a new 10 CFR 63 as discussed in Section III above. Information from DG-3003 (NRC 1990) considered likely to be applicable under the new regulation is included in the technical guidance document, though DG-3003 is not referenced as an authoritative source for such information.

In addition to the documents identified above, the information in the *Mined Geologic Disposal System License Application Annotated Outline*, YMP/94-05 (DOE Office of Civilian Radioactive Waste Management, in draft), was used extensively to develop the topics and regulatory interfaces contained herein.

9.0 PURPOSE, SUMMARY, AND CONCLUSIONS

Provide the purpose of Chapter 9 of the License Application, and relate the information in the chapter to information in Chapters 3, 4, 7, and 10. This will include a summary of the information contained in the chapter and a discussion of the organization of the chapter. Provide a summary of conclusions supporting the intent of this chapter and a description of important uncertainties, if applicable, that affect the conclusions.

Seismic Design—The Yucca Mountain Site Characterization Project has produced a *Topical Report: Preclosure Seismic Design Methodology for a Geologic Repository at Yucca Mountain* (YMP 1996) for the source of the seismic design bases. NRC has a series of regulatory guides (1.29, 1.60, 1.61, 1.92, 1.100, 1.122, 1.124, 1.130, 1.132, 1.143, and others) that guide designers

regarding classification and design of structures to resist seismic events. Unless any of these regulatory guides were used as a design basis for the radwaste structures, systems, and components, reference in this section the topical report as the source for seismic design information and methodology as applied to radwaste structures, systems, and components (see each system description section below for further direction regarding Regulatory Guide 1.143 [NRC 1979a]).

Regulatory Guides—NRC has promulgated many regulatory guidance documents that are generically applicable to design, construction, and testing. Do not list each such generally or widely applicable document that might be applicable with the discussion of each affected radwaste structure, system, or component in Chapter 9. Instead, state which of these regulatory guides apply to the design of the radwaste systems. For each, state the applicability of the regulatory guide and the extent to which the design complies with the regulatory positions in the regulatory guides. If the regulatory guide can reasonably be considered applicable but is not being used or not fully used, explain why.

The list of regulatory guides below should be considered for applicability to the radwaste structures, systems, and components as described in the previous paragraphs. Review these documents; some of their guidance is potentially applicable. The project will assess the applicability of the guidance in these documents and appropriate instructions for the License Application author will be provided in a future revision of the technical guidance document.

- Regulatory Guide 1.31, *Control of Ferrite Content in Stainless Steel Weld Metal* (NRC 1977a), describes a method acceptable to NRC for implementing requirements with regard to the control of welding in fabricating and joining safety-related austenitic stainless steel components and systems.

- Regulatory Guide 1.34, *Control of Electroslag Weld Properties* (NRC 1972), describes an acceptable method of implementing these requirements with regard to the control of weld properties when fabricating electroslag welds for nuclear components made of ferritic or austenitic materials.

- Regulatory Guide 1.36, *Nonmetallic Thermal Insulation for Austenitic Stainless Steel* (NRC 1973g), contains guidance on insulation specifications and fluid ion content for stainless piping.

- Regulatory Guide 1.38, *Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants* (NRC 1977c), describes protection of stainless parts from packaging contamination.

- Regulatory Guide 1.39, *Housekeeping Requirements for Water-Cooled Nuclear Power Plants* (NRC 1977b), describes housekeeping requirements during construction.

- Regulatory Guide 1.43, *Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components* (NRC 1973c), describes controls for underclad weld cracking when cladding low-alloy steel with stainless steel.

- Regulatory Guide 1.44, *Control of the Use of Sensitized Stainless Steel* (NRC 1973d), describes controls to prevent stress corrosion cracking.

- Regulatory Guide 1.50, *Control of Preheat Temperature for Welding of Low-Alloy Steel* (NRC 1973b), describes an acceptable method of implementing General Design Criterion 1, "Quality Standards and Records," of Appendices A and B to 10 CFR 50 requirements with regard to the control of welding for low-alloy steel components during initial fabrication.

- Regulatory Guide 1.54, *Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants* (NRC 1973h), specifies procedures for application of protective coatings to a variety of materials.

- Regulatory Guide 1.116, *Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment* (NRC 1976d), refers to ANSI N45.2.8-1975 (ANSI 1975) for installation, inspection, and testing of equipment.

- Regulatory Guide 1.136, *Materials, Construction, and Testing of Concrete Containments* (NRC 1981b), describes materials control and control of special processes, such as welding, and proper testing to be performed with regard to the materials, construction, and testing of concrete containments.

- Regulatory Guide 1.148, *Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants* (NRC 1981a), specifies information pertinent to defining operating requirements for valve assemblies whose safety-related function is to open, close, or regulate fluid flow.

- Regulatory Guide 3.30, *Selection, Application, and Inspection of Protective Coatings (Paints) for Fuel Reprocessing Plants* (NRC 1975c), describes practices for the selection, application, and inspection of protective coatings.

- Regulatory Guide 3.37, *Guidance for Avoiding Intergranular Corrosion and Stress Corrosion in Austenitic Stainless Steel* (NRC 1975b), specifies procedures for controlling the use and testing of austenitic stainless steels to avoid intergranular corrosion and stress corrosion.

- Regulatory Guide 5.42, *Design Considerations for Minimizing Residual Holdup of Special Nuclear Material in Equipment for Dry Process Operations* (NRC 1975a), describes design features for minimizing the residual holdup of special nuclear material after draindown or cleanout of equipment used in dry process operations.

Design Input Data–In appropriate locations in this chapter, for any computer codes discussed, identify the computer code and the verification and validation status of the computer code. Identify the data which require qualification. Identify the extent of compliance of any peer review and expert elicitation results with the requirements of DOE-RW/0333P, *Quality Assurance Requirements and Description* (DOE 1998).

9.1 SOURCE TERMS

The purpose of this section is to characterize the radioactive wastes generated during repository operations and those released both within the geologic repository operations area and to the environment. This characterization will support the dose assessments described in Chapters 7 and 10 of the License Application.

Describe the characteristics of the radioactive wastes associated with repository normal operations, anticipated operational occurrences, and accident situations, including radioactive effluents. Provide a summary that identifies each effluent and the type of waste. Include effluents from the following waste types: treated process effluent from waste treatment areas, sewage, drinking water, rain runoff, laundry waste, and items requiring further development. Identify the radionuclides expected in various portions of the radioactive waste management systems, including tanks, ion exchangers, evaporators, filters, and crud traps, as well as those released during venting operations. Identify the amount of effluent generated per metric ton (or other unit) of waste handled and stored per unit of time. Include in the characterization the sources, chemical and physical form, particle sizes (AMAD), projected activities/concentrations, lung solubility classes, and the total quantity and rates, such as annual, peak, per waste package, per occurrence of each radionuclide discharged per unit time in each stream. Present the data by waste type that is liquid, solid, gaseous, and mixed low-level radioactive waste in the appropriate subsection. Include bases or assumptions to substantiate the discussions. Identify mathematical models used to calculate source terms where applicable.

9.1.1 Liquid Low-Level Radioactive Waste Source Terms

Provide characteristics for the liquid low-level radioactive waste generated from equipment and facility decontamination operations, such as facility floor drains, cask/canister/container decontamination, and tooling and component decontamination. Identify expected concentrations in various portions of the liquid low-level radioactive waste system for normal operations, anticipated operational occurrences, and accident situations. Identify sources of leakage, and provide estimates of and bases for the activity predicted to become airborne from these sources. Identify potential sources of leakage to the environment during normal operations, anticipated operational occurrences, and accident situations.

9.1.2 Solid Low-Level Radioactive Waste Source Terms

Provide characteristics for the solid low-level radioactive waste expected to be generated from waste management and decontamination operations, housekeeping activities, and maintenance activities conducted within the radiologically controlled areas, such as swipes, coveralls, spent resin slurries, evaporator bottoms, contaminated components, and high-efficiency particulate air filters. Tabulate the following: maximum and expected waste inputs; physical forms, such as resin, sludge, filters, and tools; sources of waste; volume per batch; isotopic composition; and activity levels expected. Provide the bases for the values provided.

9.1.3 Gaseous Waste Source Terms

Provide characteristics for the radioactive gases and airborne particulates expected to be generated from waste handling and treatment operations, such as cask venting, spent fuel assembly handling, and low-level radioactive waste treatment equipment vents. Provide the characteristics for gaseous and particulate radioactive effluents for normal operations, anticipated operational occurrences, and accident situations. Identify sources of leakage, and provide estimates of and bases for the activity predicted from these sources. Identify sources and transport paths from systems not normally part of the radioactive waste management systems but which may be sources under anticipated operational occurrences or accident situations.

9.1.4 Mixed Low-Level Radioactive Waste Source Terms

Provide characteristics for mixed radioactive waste expected to be generated for the operational conditions identified above.

9.2 LIQUID LOW-LEVEL RADIOACTIVE WASTE MANAGEMENT SYSTEM

The liquid low-level radioactive waste management system may be addressed as a whole or may be divided into subsystems for discussion. If the system is divided into subsystems, organize this section so that each of the subsystems is identified under a section, such as Section 9.2.1, with each of the following headings (design bases, system description, and design evaluation) appearing for each subsystem and numbered with an appropriate section designation, such as 9.2.1.1. Under each section, further subdivisions may be created at the discretion of the author, particularly if organization by discipline or component is useful for the clarity of the system discussion. Combine the discussion of potential releases from identified subsystems into one section, Section 9.2.3.

9.2.1 Design Bases

10 CFR 60.130 requires that a License Application to receive, possess, store, and dispose of high-level radioactive waste in the geologic repository operations area must include the principal design criteria for the facility. 10 CFR 60.21(c)(2) requires that the Safety Analysis Report includes a description of the design criteria and their relationship to general performance objectives of NRC, design bases and their relation to the principal design criteria, information relative to materials of construction, and codes and standards that will be applied to the geologic repository operations area. Describe the design bases[2] for the liquid low-level radioactive waste

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[2] 10 CFR 60 definition: "Design bases" means that information that identifies the specific functions to be performed by a structure, system, or component 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 a structure, system, or component must meet its functional goals. The values for controlling parameters for external events include: (1) 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; and (2) estimates

management system/subsystems. Include the bases for the systems installed to manage and contain liquid effluents that are expected to be discharged into the underground or surface facility as a result of normal operations, anticipated operational occurrences, and accident situations. Identify the relationship of the bases to geologic repository operations area performance objectives.

In the design bases description, include information that satisfies each of the following references to the CFR. See Section III for the full text of the CFR requirement.

- 10 CFR 60.111(a) requires that the geologic repository operations area be designed so that, until permanent closure, radiation exposures, radiation levels, and releases of radioactive materials to unrestricted areas will be maintained within the limits specified in 10 CFR 20 and such generally applicable environmental standards for radioactivity as may have been established by U.S. Environmental Protection Agency (EPA).

- 10 CFR 60.131(a)(1), (2), (3), (4), and (6) require that the geologic repository operations area be designed to maintain doses, levels, and concentrations of radioactive material in air in restricted areas within the limits specified in 10 CFR 20. Address the specific requirements of those 10 CFR 60 references.

- 10 CFR 60.131(b) requires that structures, systems, and components important to safety be designed so that they will perform their necessary safety functions, assuming occurrence of design basis events.

- 10 CFR 60.131(c) requires that structures, systems, and components important to safety be designed to withstand dynamic effects, such as missile impacts, that could result from equipment failure, and similar events and conditions that could lead to loss of their safety functions.

- 10 CFR 60.131(d)(1), (2), and (4) require implementation of specified fire and explosion protection considerations for structures, systems, and components important to safety.

- 10 CFR 60.131(e)(1) requires that structures, systems, and components important to safety be designed to maintain control of radioactive waste and radioactive effluents and to permit prompt termination of operations during an emergency.

- 10 CFR 60.131(f)(3) requires that provisions be made that, on loss of primary electric power source or circuit, reliable and timely emergency power can be provided to instruments and operating systems, including alarm systems, important to safety.

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of severe external man-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.

- 10 CFR 60.131(g) requires that structures, systems, and components important to safety be designed to permit periodic inspection, testing, and maintenance as necessary to ensure their continued functioning and readiness.

- 10 CFR 60.131(i) requires that the design include provisions for instrumentation and control systems to monitor and control the behavior of systems important to safety, assuming occurrence of design basis events.

- 10 CFR 60.132(c)(1) requires that the surface facilities be designed to control the release of radioactive materials in effluents during Category 1 design basis events so as to meet the performance objectives of Section 60.111(a).

- 10 CFR 60.132(d) requires that radioactive waste treatment facilities be designed to process any radioactive wastes generated at the geologic repository operations area into a form suitable to permit safe disposal at the geologic repository operations area or to permit safe transportation and conversion to a form suitable for disposal at an alternative site in accordance with any regulations that are applicable.

- 10 CFR 60.132(e) requires that the surface facility be designed to facilitate decontamination or dismantlement to the same extent as would be required, under other parts of that chapter, with respect to equivalent activities licensed thereunder.

- 10 CFR 60.136(b) requires that, for Category 2 design basis events, no individual located on or beyond any point on the boundary of the preclosure controlled area will receive the more limiting of a total effective dose equivalent to 0.05 Sv (5 rem), or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.5 Sv (50 rem). The eye dose equivalent shall not exceed 0.15 Sv (15 rem), and the shallow dose equivalent to skin shall not exceed 0.5 Sv (50 rem).

9.2.2 System Description

Introduce this section with a description of the liquid low-level radioactive waste management philosophy. Provide a general design description, referring to site and location/building maps to show how the system relates to and works with other systems.

Provide design criteria[3] for the liquid low-level radioactive waste management systems. Design criteria are to be obtained from project documents, such as system description documents.

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[3] 10 CFR 60.130 states, "The principal design criteria establish the necessary design, fabrication, construction, testing, maintenance, and performance requirements for structures, systems, and components important to safety and/or important to waste isolation. Sections 60.131 through 60.134 specify minimum requirements for the principal design criteria for the geologic repository operations area."

General Design Criteria—Describe the general design criteria that govern the design of the liquid low-level radioactive waste management system using the technical criteria from Subpart E to 10 CFR 60 and using a similar design philosophy to that contained in the reactor general design criteria of 10 CFR 50, Appendix A. Such criteria should relate to topics such as those that follow:

- **Quality Standards and Records**—State (or refer to another location in this chapter that identifies) the major codes and standards to which the system will be designed, fabricated, inspected, and tested to help ensure integrity and high quality; and provide a brief justification for those chosen. Identify which system components will be controlled in accordance with the requirements of DOE/RW-0333P, *Quality Assurance Requirements and Description* (DOE 1998). State that records will be kept for design, construction, and testing of the system and its components. Provide a cross-reference to the appropriate procedure or process that will be used to ensure these records are made and retained. Refer the reader to Chapter 14 of the License Application for additional information on the quality assurance program.

- **Design Bases for Protection Against Natural Phenomena**—State that the system is designed and will be constructed to withstand the effects of earthquakes and other design natural phenomena (list them) without loss of capability to perform its safety functions. Describe briefly the major design features that provide assurance of protection.

- **Fire Protection**—State that the system is designed and will be constructed to minimize the probability and effect of fires and explosions. Briefly describe the design features that provide this assurance.

- **Environmental and Dynamic Effects Design Basis**—State that the system is designed and will be constructed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. Describe briefly the design features that provide this assurance. Mention the dynamic effects (such as missiles, pipe whipping, and discharging fluids) against which the system is protected. Refer to any information elsewhere in the License Application that supports this discussion.

- **Control of Releases of Radioactive Materials to the Environment**—State that the system is designed to control the release of liquid radioactive material. Describe the means to control the release of liquid effluents and to handle radioactive liquid wastes produced during normal operation, including anticipated operational occurrences. Include a description of holdup capacity that is provided for retention of liquid waste containing radioactive materials.

- **Radioactivity Control**—State that the system is designed to ensure adequate safety under normal and postulated accident conditions. Briefly describe the design provisions for periodic inspection and testing of components important to safety and for suitable shielding for radiation protection.

For the system or each of the subsystems, include the following (tabulated information is encouraged where appropriate):

- Purpose of the system.

- A description of the design and performance requirements for structures, systems, and components of the geologic repository that are important to safety (10 CFR 60.21(c)(3)) and a statement that the system meets these performance requirements. Refer to Chapter 7 for detailed analysis and demonstration that the design and performance requirements for Category I and II design basis events have been met.

- Safety classification of the system/subsystem and major components and the rationale for that classification. Refer the License Application reader to Chapter 2 for detailed discussion of the classification process. For major systems/components, include analysis of system/component performance when identifying those important to safety. For the purposes of this analysis, assume that operations at the geologic repository operations area will be carried out at the maximum capacity and rate of receipt of radioactive waste (10 CFR 60.21(c)(1)(ii)(E)). For liquid low-level radioactive waste management system components, list and describe key design parameters, such as duty, capacity, horsepower, safety features, flow, temperature, pressure, and materials of construction.

- Each radioactive liquid low-level radioactive waste source and the volume of waste anticipated from each source.

- For each system/subsystem, a description of how the system/subsystem achieves its functions and design bases; flow paths; major components; materials; capacities; collection; temporary storage capabilities, including storage method, capacities, and related facilities; treatment; packaging; method(s) of disposal; and system/subsystem operations and protective actions, including those intended to prevent release of liquid waste to unrestricted areas. For each effluent, describe the constraints imposed on process systems and equipment to ensure meeting the performance objectives in 10 CFR 60.111(a). If the methods of disposal include shipment to a burial facility, make a statement that the facility will have the licenses or authorizations required by the governing authority for receipt of the shipped waste.

- System/subsystem response to normal operations, anticipated operational occurrences, and accident situations.

- Electrical power requirements, including redundancy.

- Instrumentation and control systems associated with the system/subsystem.

- Interfaces with other systems/subsystems.

- Applicable codes and standards.

- Design alternatives that were considered and the rationale for selecting the current design, as applicable.

- Design features and procedures that minimize generation of liquid waste and provide for the minimization of the possibility of spread of contamination and for control and containment of spills.

- ALARA considerations incorporated into the design, including cost/benefit analyses that were performed.

- How the system relates to the overall repository performance or to the related NRC regulations.

- A description of system design considerations, if any, that are intended to facilitate system or facility decontamination or dismantlement (10 CFR 60.21(c)(11)).

Provide system diagrams to support and further explain the system description(s). On the drawings, identify and show any radiological design features, such as shielding and thicknesses.

10 CFR 60.131(h) requires the following:

- All systems for processing, transporting, handling, storage, retrieval, emplacement, and isolation of radioactive waste must be designed to ensure that nuclear criticality is not possible unless at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety.

- Each system must be designed for criticality safety assuming occurrence of design basis events.

- The calculated effective multiplication factor (k_{eff}) must be sufficiently below unity to show at least a five percent margin, after allowance for the bias in the method of calculation and the uncertainty in the experiments used to validate the method of calculation.

Address these requirements for the liquid low-level radioactive waste management system(s), either providing detailed information that these CFR requirements are met or justifying that criticality in the system is not a credible event. Cross-reference Chapter 7 information for design basis events involving accidents.

Regulatory Guide 1.143, *Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants* (NRC 1979a), furnishes design guidance acceptable to NRC relating to seismic and quality group classification and quality assurance provisions for radioactive waste management structures, systems, and components. It also describes provisions for controlling releases of liquids containing radioactive materials, such as spills or tank overflows, from all systems. Regulatory Position 1 of this regulatory guide addresses systems handling radioactive materials in liquids; Regulatory Position 4 identifies additional design, construction, and testing criteria; Regulatory

Position 5 addresses seismic design for radioactive waste management systems and structures housing radioactive waste management systems; and Regulatory Position 6 identifies quality assurance for radioactive waste management systems. Review this guide. Some of the guidance is potentially useful and applicable. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of this technical guidance document.

10 CFR 20.1101(b) requires that the licensee use, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are ALARA. Regulatory Guide 8.8, *Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable* (NRC 1978f), provides information relevant to attaining ALARA goals and objectives. Regulatory Position 2, Facility and Equipment Design Features, of that regulatory guide contains practices for incorporating ALARA principles into the design of facilities. Make a statement that indicates the extent (if any) that the geologic repository operations area design is consistent/compliant with these aspects of regulatory guidance. Describe procedures, engineering controls, and specific design features that apply to the liquid low-level radioactive waste system/subsystems that will be incorporated to apply ALARA principles to those systems. Relate specific design features to those discussed in Regulatory Position 2 of Regulatory Guide 8.8, as applicable. Identify ALARA features that apply to dose reduction both of the public and of the occupational worker. Demonstrate that for the liquid low-level radioactive waste management system, a reasonable effort has been made to maintain releases of radioactive material in effluents to unrestricted areas (and associated exposures) ALARA.

If the system/subsystem design includes concrete radiation shields, Regulatory Guide 1.69, *Concrete Radiation Shields for Nuclear Power Plants* (NRC 1973a), which implements ANSI N101.6-1972, *Concrete Radiation Shields* (ANSI 1972), with specified exceptions, may provide useful and applicable information. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of this technical guidance document.

Cost-benefit analyses are considered by NRC to be an integral part of a viable ALARA program. Lack of cost-benefit analyses related to ALARA design features are considered to be de facto evidence that the ALARA program is not properly implemented. Describe in this section any cost-benefit analyses associated with the design of the liquid low-level radioactive waste system/subsystems. Regulatory Guide 1.110, *Cost-Benefit Analysis for Radioactive Waste Systems for Light-Water-Cooled Nuclear Power Reactors* (NRC 1976c), provides guidance on acceptable practices for cost-benefit analyses that involve consideration of reduction in dose to the general public. Review this guide. Some of the guidance is potentially useful and applicable. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of this technical guidance document.

9.2.3 Design Evaluation

For each leakage source identified in Section 9.1 for the liquid low-level radioactive waste management system(s), identify the expected releases of radioactive liquids, the bases for the values used, and release paths during normal operations, anticipated operational occurrences, and accident situations.[4] Identify the parameters and assumptions used for the release calculations. Tabulate the releases by radionuclide for each subsystem identified and for the total system and indicate effluent concentrations. Compare the calculated effluents with the concentration limits of 10 CFR 20, Appendix B, Table II, Column 2, and with EPA preclosure dose limits (if they are promulgated) (see 10 CFR 60.111). With references to Chapter 7, compare the doses due to the effluents with the numerical design objectives of 10 CFR 50, Appendix I, and to the system design bases. Identify dilution factors considered in the evaluation.

9.3 SOLID LOW-LEVEL WASTE MANAGEMENT SYSTEM

The solid low-level radioactive waste management system may be addressed as a whole or may be divided into subsystems for discussion. If the system is divided into subsystems, organize this section so that each of the subsystems is identified under a section, such as 9.3.1, with each of the following headings (design bases, system description, and design evaluation) appearing for each subsystem and numbered with an appropriate section designation, such as 9.3.1.1. Under each section, further subdivisions may be created at the discretion of the author if needed, particularly if organization by discipline or component is useful for the clarity of the system discussion. Combine the discussion of potential releases from identified subsystems into one section, Section 9.3.3.

9.3.1 Design Bases

10 CFR 60.130 requires that an application to receive, possess, store, and dispose of high-level radioactive waste in the geologic repository operations area must include the principal design criteria for a proposed facility. 10 CFR 60.21(c)(2) requires that the Safety Analysis Report includes a description of the design criteria and their relationship to general performance objectives of NRC, design bases and their relation to the principal design criteria, information relative to materials for construction, and codes and standards that will be applied to the geologic repository operations area. Describe the design bases for the solid low-level radioactive waste management system/subsystems, including the wet solid low-level radioactive waste subsystem to be used for processing ion exchange resins, filter sludges, evaporator bottoms, and miscellaneous liquids. Include the bases for the systems installed to manage (including processing for disposal) solid low-level radioactive waste materials from the underground or surface facility as a result of normal operations, anticipated operational occurrences, and accident situations. Identify the relationship of the bases to geologic repository operations area performance objectives.

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[4] Dose assessments to the public associated with releases to the environment are provided in Chapter 7 and are not required, but may be referenced, in this chapter.

In the design bases description, include information that satisfies each of the following references to the CFR. See Section III for the full text of the CFR requirement.

- 10 CFR 60.111(a) requires that the geologic repository operations area be designed so that, until permanent closure, radiation exposures, radiation levels, and releases of radioactive materials to unrestricted areas will be maintained within the limits specified in 10 CFR 20 and such generally applicable environmental standards for radioactivity as may have been established by EPA.

- 10 CFR 60.131(a)(1), (2), (3), (4), and (6) require that the geologic repository operations area be designed to maintain doses, levels, and concentrations of radioactive material in air in restricted areas within the limits specified in 10 CFR 20, and address specific requirements.

- 10 CFR 60.131(b) requires that structures, systems, and components important to safety be designed so that they will perform necessary safety functions, assuming occurrence of design basis events.

- 10 CFR 60.131(c) requires that structures, systems, and components important to safety be designed to withstand dynamic effects, such as missile impacts, that could result from equipment failure, and similar events and conditions that could lead to loss of safety functions.

- 10 CFR 60.131(d)(1), (2), and (4) require implementation of specified fire and explosion protection considerations for structures, systems, and components important to safety.

- 10 CFR 60.131(e)(1) requires that structures, systems, and components important to safety be designed to maintain control of radioactive waste and radioactive effluents and to permit prompt termination of operations during an emergency.

- 10 CFR 60.131(f)(3) requires that provisions be made that, on loss of primary electric power source or circuit, reliable and timely emergency power can be provided to instruments and operating systems, including alarm systems, important to safety.

- 10 CFR 60.131(g) requires that structures, systems, and components important to safety be designed to permit periodic inspection, testing, and maintenance as necessary to ensure their continued functioning and readiness.

- 10 CFR 60.131(i) requires that the design include provisions for instrumentation and control systems to monitor and control the behavior of systems important to safety, assuming occurrence of design basis events.

- 10 CFR 60.132(c)(1) requires that the surface facilities be designed to control the release of radioactive materials in effluents during Category 1 design basis events so as to meet the performance objectives of Section 60.111(a).

- 10 CFR 60.132(d) requires that radioactive waste treatment facilities be designed to process any radioactive wastes generated at the geologic repository operations area into a form suitable to permit safe disposal at the geologic repository operations area or to permit safe transportation and conversion to a form suitable for disposal at an alternative site in accordance with any regulations that are applicable.

- 10 CFR 60.132(e) requires that the surface facility be designed to facilitate decontamination or dismantlement to the same extent as would be required, under other parts of that chapter, with respect to equivalent activities licensed thereunder.

- 10 CFR 60.136(b) requires that for Category 2 design basis events, no individual located on or beyond any point on the boundary of the preclosure controlled area will receive the more limiting of a total effective dose equivalent to 0.05 Sv (5 rem), or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) to 0.5 Sv (50 rem). The eye dose equivalent shall not exceed 0.15 Sv (15 rem), and the shallow dose equivalent to skin shall not exceed 0.5 Sv (50 rem).

9.3.2 System Description

This section will provide descriptions of the onsite waste management systems used to collect, process, and package solid low-level radioactive waste materials that are generated at the site.

Provide design criteria for the solid low-level radioactive waste management system. Obtain these criteria from project documents, such as system description documents.

General Design Criteria—Describe the general design criteria that govern the design of the solid liquid low-level radioactive waste management system using the technical criteria from Subpart E to 10 CFR 60 and using a similar design philosophy to that contained in the reactor general design criteria of 10 CFR 50, Appendix A. Such criteria should relate to topics such as those that follow:

- **Quality Standards and Records**—State (or refer to another location in this chapter that identifies) the major codes and standards to which the system will be designed, fabricated, inspected, and tested to help ensure integrity and high quality; and provide a brief justification for those chosen. Identify which system components will be controlled in accordance with the requirements of DOE/RW-0333P, *Quality Assurance Requirements and Description* (DOE 1998). State that records will be kept for design, construction, and testing of the system and its components; provide a cross-reference to the appropriate procedure or process that will be used to ensure these records are made and retained. Refer the reader to Chapter 14 of the License Application for additional information on the quality assurance program.

- **Design Bases for Protection Against Natural Phenomena**—State that the system is designed and will be constructed to withstand the effects of earthquakes and other design natural phenomena (list them) without loss of capability to perform its safety

functions. Describe briefly the major design features that provide assurance of protection.

- **Fire Protection**—State that the system is designed and will be constructed to minimize the probability and effect of fires and explosions. Briefly describe the design features that provide this assurance.

- **Environmental and Dynamic Effects Design Basis**—State that the system is designed and will be constructed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. Describe briefly the design features that provide this assurance. Mention the dynamic effects (such as missiles, pipe whipping, and discharging fluids) against which the system is protected. Refer to any information elsewhere in the License Application that supports this discussion.

- **Control of Releases of Radioactive Materials to the Environment**—State that the system is designed to handle solid low-level radioactive wastes. Describe the means to handle solid low-level radioactive wastes produced during normal repository operation, including anticipated operational occurrences.

- **Radioactivity Control**—State that the system is designed to ensure adequate safety under normal and postulated accident conditions. Briefly describe the design provisions for periodic inspection and testing of components important to safety and for suitable shielding for radiation protection.

For the system or each of the subsystems, include the following information (tabulated information is encouraged where appropriate):

- Purpose of the system/subsystem.

- A description of the design and performance requirements for structures, systems, and components of the geologic repository that are important to safety (10 CFR 60.21(c)(3)), and a statement that the system meets these performance requirements. Refer to Chapter 7 for detailed analysis and demonstration that the design and performance requirements for Category I and II design basis events have been met.

- Safety classification of the system/subsystem and major components and the rationale for that classification. Refer the License Application reader to Chapter 2 for detailed discussion of the classification process. For major systems/components, include analysis of system/component performance when identifying those important to safety. For the purposes of this analysis, assume that operations at the geologic repository operations area will be carried out at the maximum capacity and rate of receipt of radioactive waste (10 CFR 60.21(c)(1)(ii)(E)). For solid low-level radioactive waste management system components, list and describe key design parameters, such as duty, capacity, horsepower, safety features, flow, temperature, pressure, and materials of construction.

- Each solid low-level radioactive waste source and the volume of waste anticipated from each source.

- For each system/subsystem, a description of how the system/subsystem achieves its design bases and functions related to wet and dry solid radioactive waste; handling capabilities; major components; materials; maximum and expected capacities; collection; temporary storage capabilities, including storage method, capacities, and related facilities; treatment; packaging; method of disposal; and system/subsystem operations and protective actions, including design/operational/protective features intended to prevent release of radioactive waste to unrestricted areas due to overflows from tanks containing liquids, sludges, and spent resins, such as the effectiveness of physical and monitoring precautions taken, such as retention basins, curbing, and level gauges. If the methods of disposal include shipment to a burial facility, make a statement that the facility will have the licenses or authorizations required by the governing authority for receipt of the shipped waste.

- Method to be used for solidifying each waste type, the type of container to be used for packaging, and the means to be used to ensure the absence of free liquid in the waste containers and to ensure a solid matrix.

- Provisions to be used to control airborne radioactivity due to dust during compaction and baling, containers to be used for packaging dry waste and their conformance with applicable federal regulations, provisions for decontaminating and moving containers to shipping areas, and potential for radioactive spills due to dropping packages.

- System/subsystem's response, if any, to normal operations, anticipated operational occurrences, and accident situations.

- Electrical power requirements, including redundancy.

- Instrumentation and control systems such as those used for process control associated with the system/subsystem.

- Interfaces with other systems/subsystems, including tanks and equipment that use compressed gases for any function (provide information on gas flow rates, volume per operation, expected number of operations per year, and expected radionuclide concentration of offgases, treatment provided, and interfaces with ventilation exhaust systems).

- Applicable codes and standards.

- Design alternatives that were considered and the rationale for selecting the current design (as applicable).

- Design features and procedures that minimize generation of solid radioactive waste.

- ALARA considerations incorporated into the design, including cost/benefit analyses that were performed.

- Relationship of the system to the overall repository performance or to the related NRC regulations.

- A description of system design considerations, if any, that are intended to facilitate system or facility decontamination or dismantlement (10 CFR 60.21(c)(11)).

Provide system diagrams to support the system description(s) and layout drawings of the packaging, storage, and shipping areas. On the drawings, identify and show any radiological design features, such as shielding and thicknesses. Tabulate the maximum and expected annual volumes and curie and isotopic content of wastes to be shipped offsite for each waste category, using references to Section 9.1.2, as applicable.

Regulatory Guide 1.143, *Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants* (NRC 1979a), furnishes design guidance acceptable to NRC relating to seismic and quality group classification and quality assurance provisions for radioactive waste management structures, systems, and components. It also describes provisions for controlling releases of liquids containing radioactive materials, such as spills or tank overflows, from all systems. Regulatory Position 3 of this guide addresses solid radioactive waste systems, including those used for waste solidification, slurry collection and settling, and separating; Regulatory Position 4 identifies additional design, construction, and testing criteria; Regulatory Position 5 addresses seismic design for radioactive waste management systems and structures housing radioactive waste management systems; and Regulatory Position 6 identifies quality assurance for radioactive waste management systems. Review this guide. Some of the guidance is potentially useful and applicable. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of this technical guidance document.

10 CFR 20.1101(b) requires that the licensee use, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are ALARA. Regulatory Guide 8.8, *Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable* (NRC 1978f), provides information relevant to attaining ALARA goals and objectives. Regulatory Position 2, Facility and Equipment Design Features, contains practices for incorporating ALARA principles into the design of facilities. State the extent (if any) that the geologic repository operations area design is consistent/compliant with these aspects of regulatory guidance. Describe procedures, engineering controls, and specific design features that apply to the solid low-level radioactive waste management system/subsystems that will be incorporated to apply ALARA principles to those systems. Relate specific design features to those discussed in Regulatory Position 2 of Regulatory Guide 8.8 (NRC 1978f), as applicable. Identify ALARA features that apply to dose reduction both of the public and of the occupational worker.

If the system/subsystem design includes concrete radiation shields, Regulatory Guide 1.69, *Concrete Radiation Shields for Nuclear Power Plants* (NRC 1973a), which implements ANSI N101.6-1972, *Concrete Radiation Shields* (ANSI 1972), with specified exceptions, may provide useful and applicable information. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of this technical guidance document.

Cost-benefit analyses are considered by NRC to be an integral part of a viable ALARA program. Lack of cost-benefit analyses related to ALARA design features are considered to be de facto evidence that the ALARA program is not properly implemented. Describe in this section any cost-benefit analyses associated with the design of the gaseous low-level radioactive waste system/subsystems. Regulatory Guide 1.110, *Cost-Benefit Analysis for Radioactive Waste Systems for Light-Water-Cooled Nuclear Power Reactors* (NRC 1976c), provides guidance on acceptable practices for cost-benefit analyses that involve consideration of reduction in dose to the general public. Review this guide. Some of the guidance is potentially useful and applicable. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of this technical guidance document.

9.3.3 Design Evaluation

For each source identified in Section 9.1 for the solid low-level radioactive waste management system, identify the expected releases of radioactive solids, the bases for the values used, and release paths during normal operations, anticipated operational occurrences, and accident situations.[5] Identify the parameters and assumptions used for the release calculations. Tabulate the releases by radionuclide for each subsystem identified above and for the total system and indicate effluent concentrations. With references to Chapter 7, compare the doses due to the effluents with the numerical design objectives of 10 CFR 50, Appendix I, and to the system design bases (specifically to those associated with dose limits of 10 CFR 20 and to EPA preclosure dose limits (if they are promulgated)) (see 10 CFR 60.111). If no solid radioactive releases are anticipated under any or all of these operational conditions, then make a statement to that effect and explain.

9.4 GASEOUS WASTE MANAGEMENT SYSTEM

The gaseous low-level radioactive waste management system may be addressed as a whole or may be divided into subsystems for discussion. If the system is divided into subsystems, organize this section so that each of the subsystems is identified under a subsection, such as 9.4.1, with each of the following headings (design bases, system description, and design evaluation) appearing for each subsystem and numbered with an appropriate subsection designation, such as 9.4.1.1. Under each subsection, further subdivisions may be created at the discretion of the author if needed, particularly if organization by discipline or component is

[5] Dose assessments to the public associated with releases to the environment are provided in Chapter 7 and are not required, but may be referenced, in this chapter.

useful for the clarity of the system discussion. Combine the discussion of potential releases from identified subsystems into one section, Section 9.4.3.

9.4.1 Design Bases

10 CFR 60.130 requires that the License Application to receive, possess, store, and dispose of high-level radioactive waste in the geologic repository operations area must include the principal design criteria for a proposed facility. 10 CFR 60.21(c)(2) requires that the Safety Analysis Report include a description of the design criteria and their relationship to general performance objectives of NRC, design bases and their relation to the principal design criteria, information relative to materials of construction, and codes and standards that will be applied to the geologic repository operations area. Describe the design bases for the gaseous low-level radioactive waste management system/subsystems. Include the bases for the systems installed to manage, contain, and release gaseous and particulate effluents that are expected to be discharged into the underground or surface facility as a result of normal operations, anticipated operational occurrences, and accident situations. Identify the relationship of the bases to geologic repository operations area performance objectives.

In the design bases description, include information that satisfies each of the following references to the CFR. See Section III of this chapter for the full text of the CFR requirement.

- 10 CFR 20.1301 specifies dose limits to members of the public that result from geologic repository operations area operations.

- 10 CFR 20.1101(d) requires that the licensee, as part of the requirements for implementing Subsection 2011.01(b), constrain air emissions to limit doses from those emissions. These constraints apply in addition to the requirements of Section 20.1301.

- 10 CFR 60.111(a) requires that the geologic repository operations area be designed so that, until permanent closure, radiation exposures, radiation levels, and releases of radioactive materials to unrestricted areas will be maintained within the limits specified in 10 CFR 20 and such generally applicable environmental standards for radioactivity as may have been established by EPA.

- 10 CFR 60.131(a)(1), (2), (3), (4), and (6) require that the geologic repository operations area be designed to maintain doses, levels, and concentrations of radioactive material in air in restricted areas within the limits specified in 10 CFR 20, and address specific requirements.

- 10 CFR 60.131(b) requires that structures, systems, and components important to safety be designed so that they will perform their necessary safety functions, assuming occurrence of design basis events.

- 10 CFR 60.131(c) requires that structures, systems, and components important to safety be designed to withstand dynamic effects, such as missile impacts, that could

result from equipment failure, and similar events and conditions that could lead to loss of their safety functions.

- 10 CFR 60.131(d)(1), (2), and (4) require implementation of specified fire and explosion protection considerations for structures, systems, and components important to safety.

- 10 CFR 60.131(e)(1) requires that structures, systems, and components important to safety be designed to maintain control of radioactive waste and radioactive effluents and to permit prompt termination of operations during an emergency.

- 10 CFR 60.131(f)(3) requires that provisions be made that, on loss of primary electric power source or circuit, reliable and timely emergency power can be provided to instruments and operating systems, including alarm systems, important to safety.

- 10 CFR 60.131(g) requires that structures, systems, and components important to safety be designed to permit periodic inspection, testing, and maintenance as necessary to ensure continued functioning and readiness.

- 10 CFR 60.131(i) requires that the design include provisions for instrumentation and control systems to monitor and control the behavior of systems important to safety, assuming occurrence of design basis events.

- 10 CFR 60.132(c)(1) requires that the surface facilities be designed to control the release of radioactive materials in effluents during Category 1 design basis events to meet the performance objectives of Section 60.111(a).

- 10 CFR 60.132(d) requires that radioactive waste treatment facilities be designed to process any radioactive wastes generated at the geologic repository operations area into a form suitable to permit safe disposal or to permit safe transportation and conversion to a form suitable for disposal at an alternative site in accordance with any regulations that are applicable.

- 10 CFR 60.132(e) requires that the surface facility be designed to facilitate decontamination or dismantlement to the same extent as would be required, under other parts of that chapter, with respect to equivalent activities licensed thereunder.

- 10 CFR 60.136(b) requires that for Category 2 design basis events, no individual located on or beyond any point on the boundary of the preclosure controlled area will receive the more limiting of a total effective dose equivalent of 0.05 Sv (5 rem), or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.5 Sv (50 rem). The eye dose equivalent shall not exceed 0.15 Sv (15 rem), and the shallow dose equivalent to skin shall not exceed 0.5 Sv (50 rem).

9.4.2 System Description

This section will provide descriptions of the onsite waste management systems used to collect, treat, and vent radioactive gases and particulates that are generated at the site.

Provide design criteria for the gaseous low-level radioactive waste management system. Obtain these criteria from project documents, such as system description documents.

General Design Criteria—Describe the general design criteria that govern the design of the gaseous low-level radioactive waste management system using the technical criteria of Subpart E to 10 CFR 60 and using a similar design philosophy to that contained in the reactor general design criteria of 10 CFR 50, Appendix A. Such criteria should relate to topics such as those that follow:

- **Quality Standards and Records**—State (or refer to another location in this chapter that identifies) the codes and standards to which the system will be designed, fabricated, inspected, and tested to help ensure integrity and high quality; and provide a brief justification for those chosen. Identify which system components will be controlled in accordance with the requirements of DOE/RW-0333P, *Quality Assurance Requirements and Description* (DOE 1998). State that records will be kept for all design, construction, and testing of the system and its components. Provide a cross-reference to the appropriate procedure or process that will be used to ensure these records are made and retained. Refer the reader to Chapter 14 of the License Application for additional information on quality assurance.

- **Design Bases for Protection Against Natural Phenomena**—State that the system is designed and will be constructed to withstand the effects of earthquakes and other design natural phenomena (list them) without loss of capability to perform its safety functions. Describe briefly the major design features that provide assurance of protection.

- **Fire Protection**—State that the system is designed and will be constructed to minimize the probability and effect of fires and explosions. Briefly describe the design features that provide this assurance.

- **Environmental and Dynamic Effects Design Basis**—State that the system is designed and will be constructed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. Describe briefly the design features that provide this assurance. Mention the dynamic effects (such as missiles, pipe whipping, and discharging fluids) against which the system is protected. Refer to any information elsewhere in the License Application that supports this discussion.

- **Control of Releases of Radioactive Materials to the Environment**—State that the system is designed to control the release of gaseous radioactive material. Describe the means to control the release of gaseous effluents produced during normal operation, including anticipated operational occurrences. Include a description of

holdup capacity (if any) that is provided for retention of gaseous waste containing radioactive materials.

- **Radioactivity Control**—State that the system is designed to ensure adequate safety under normal and postulated accident conditions. Briefly describe the design provisions for periodic inspection and testing of components important to safety and for suitable shielding for radiation protection.

For the system or each of the subsystems, include the following information (tabulated information is encouraged where appropriate).

- Purpose of the system/subsystem.

- A description of the design and performance requirements for structures, systems, and components of the geologic repository that are important to safety (10 CFR 60.21(c)(3)) and a statement that the system meets these performance requirements. Refer to Chapter 7 for detailed analysis and demonstration that the design and performance requirements for Category I and II design basis events have been met.

- Safety classification of the system/subsystem and major components, and the rationale for that classification. Refer the License Application reader to Chapter 2 for detailed discussion of the classification process. For major systems/components, include analysis of system/component performance when identifying those important to safety. For the purposes of this analysis, assume that operations at the geologic repository operations area will be carried out at the maximum capacity and rate of receipt of radioactive waste (10 CFR 60.21(c)(1)(ii)(E)). For gaseous low-level radioactive waste management system components, list and describe key design parameters, such as duty, capacity, horsepower, safety features, flow, temperature, pressure, and materials of construction.

- Each gaseous and particulate low-level radioactive waste source and the volume of waste anticipated from each source.

- For each system/subsystem, a description of how the system/subsystem achieves its functions and design bases, flow paths, major components, materials, capacities, collection, treatment, packaging, method of disposal, and system/subsystem operations and protective actions, including those intended to prevent excessive or uncontrolled release of gaseous/air particulate radioactive waste to unrestricted areas. For each effluent, describe the constraints imposed on process systems and equipment to ensure meeting the performance objectives in 10 CFR 60.111(a).

- System/subsystem response to normal operations, anticipated operational occurrences, and accident situations.

- Electrical power requirements, including redundancy.

- Instrumentation and control systems associated with the system/subsystem.

- Interfaces with other systems/subsystems.
- Applicable codes and standards.
- Design alternatives that were considered and the rationale for selecting the current design, as applicable.
- Design features and procedures that minimize generation of gaseous waste and maintain release of gaseous radioactive materials within designated limits under normal, off-normal, and accident conditions.
- ALARA considerations incorporated into the design, including cost/benefit analyses that were performed.
- Relationship of the system to the overall repository performance or to the related NRC regulations.
- A description of system design considerations, if any, that are intended to facilitate system or facility decontamination or dismantlement (10 CFR 60.21(c)(11)).

Provide system diagrams to support the system description(s), and indicate the release points of gaseous/particulate waste to the environment on these (or other) diagrams. On the drawings, identify and show any radiological design features, such as shielding and thicknesses.

The *Mined Geologic Disposal System Advanced Conceptual Design Report* (CRWMS M&O 1996) describes an integral relationship between the gaseous low-level radioactive waste system(s) and the surface ventilation system(s). Because of this relationship, the description of the gaseous low-level radioactive waste system(s) must include discussion of the interrelationship of these systems and the dependency of the disposal of gaseous waste upon the ventilation system(s); however, there is no intent to reiterate the design of the surface ventilation system(s)–that information is contained in Chapter 4. In the discussion of the gaseous system, include information to establish the relationship with the ventilation system(s), cross-referencing information in Chapter 4 related to the design of the ventilation system(s), as needed. Describe amounts and characteristics of gases/particulates to be released to the ventilation system during various geologic repository operations area activities, such as cask venting, referencing Section 9.1, as needed. Identify the difference in operation of the ventilation system under different conditions that may affect radioactive releases, such as any automatic mode changes if high airborne activity is detected. Discuss the program for measuring the efficiency of filters and other gaseous effluent treatment devices over the lifetime of the installation. Provide criteria for changing filters. Discuss how the ventilation system design will allow filter changes to be compatible with ALARA principles. If this information is provided elsewhere, reference that section rather than providing detailed discussion.

If any provisions for holdup or storage of gaseous wastes are incorporated into the design, include a description of these capabilities, including capacities.

Regulatory Guide 1.143, *Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants* (NRC 1979a), furnishes design guidance acceptable to NRC relating to seismic and quality group classification and quality assurance provisions for radioactive waste management structures, systems, and components. It also describes provisions for controlling releases of liquids containing radioactive materials, such as spills or tank overflows, from all systems. Regulatory Position 2 of this guide addresses systems handling radioactive materials in gases; Regulatory Position 4 identifies additional design, construction, and testing criteria; Regulatory Position 5 addresses seismic design for radioactive waste management systems and structures housing radioactive waste management systems; and Regulatory Position 6 identifies quality assurance for radioactive waste management systems. Review this guide. Some of the guidance is potentially useful and applicable. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision for this technical guidance document.

Regulatory Guide 3.20, *Process Offgas Systems for Fuel Reprocessing Plants* (NRC 1974b), provides guidance relative to the design of offgas treatment systems and associated equipment. Review this guide. Some of the guidance is potentially useful and applicable. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of the technical guidance document.

10 CFR 20.1301 specifies dose limits for individual members of the public and requires that each licensee conduct operations to conform with these limits. Address system design, monitoring (reference Section 9.6), and operational considerations that will lead to conformance with this regulation.

10 CFR 20.1101(b) requires that the licensee use, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are ALARA. Regulatory Guide 8.8, *Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable* (NRC 1978f), provides information relevant to attaining ALARA goals and objectives. Regulatory Position 2, Facility and Equipment Design Features, of that regulatory guide contains practices for incorporating ALARA principles into the design of facilities. State the extent (if any) that the geologic repository operations area design is consistent/compliant with these aspects of regulatory guidance. Describe procedures, engineering controls, and specific design features that apply to the gaseous low-level radioactive waste management system/subsystems that will be incorporated to apply ALARA principles to those systems. Relate specific design features to those discussed in Regulatory Position 2 of Regulatory Guide 8.8 (NRC 1978f), as applicable. Identify ALARA features that apply to dose reduction both of the public and of the occupational worker. Demonstrate that for the liquid low-level radioactive waste management system, a reasonable effort has been made to maintain releases of radioactive material in effluents to unrestricted areas and associated exposures ALARA.

If the system/subsystem design includes concrete radiation shields, Regulatory Guide 1.69, *Concrete Radiation Shields for Nuclear Power Plants* (NRC 1973a), which implements ANSI

N101.6-1972, *Concrete Radiation Shields* (ANSI 1972), with specified exceptions, may provide useful and applicable information. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of the technical guidance document.

Cost-benefit analyses are considered by NRC to be an integral part of a viable ALARA program. Lack of cost-benefit analyses related to ALARA design features is considered to be de facto evidence that the ALARA program is not properly implemented. Describe in this section cost-benefit analyses associated with the design of the gaseous low-level radioactive waste system/subsystems. Regulatory Guide 1.110, *Cost-Benefit Analysis for Radioactive Waste Systems for Light-Water-Cooled Nuclear Power Reactors* (NRC 1976c), provides guidance on acceptable practices for cost-benefit analyses that involve consideration of reduction in dose to the general public. Review this guide. Some of the guidance is potentially useful and applicable. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of the technical guidance document.

9.4.3 Design Evaluation

For each leakage source identified in Section 9.1 for the gaseous low-level radioactive waste management system, identify the expected releases of airborne radioactive gases and particulates, the bases for the values used, and release paths during normal operations, anticipated operational occurrences, and accident situations.[6] Provide the criteria to be used for releasing gaseous and air particulate wastes and the acceptable release rates. Identify the parameters and assumptions used for the release calculations. Tabulate the releases by radionuclide for each subsystem identified above and for the total system and indicate effluent concentrations. If characteristics of the effluent are not clear by reference to Section 9.1.3, provide additional information here. Compare the calculated effluents with the concentration limits of 10 CFR 20, Appendix B, Table II, Column 1, and with EPA preclosure dose limits (if they are promulgated) (see 10 CFR 60.111). With references to Chapter 7, compare the doses due to the effluents with the numerical design objectives of 10 CFR 50, Appendix I, and to the system design bases. This dose comparison should address dose contribution from gaseous/particulate waste as well as a total dose from all sources. Identify dilution factors considered in the evaluation.

9.5 MIXED WASTE MANAGEMENT SYSTEM

The mixed low-level radioactive waste management system may be addressed as a whole or may be divided into subsystems for discussion of each topic. If the system is divided into subsystems, organize this section so that each of the subsystems is identified under a subsection, such as 9.5.1, with each of the following headings (design bases, system description, and design

[6] Dose assessments to the public associated with releases to the environment are provided in Chapter 7 and are not required, but may be referenced, in this chapter. Dose assessments to occupational workers based on gaseous/particulate effluent releases are provided in Chapter 10.

evaluation) appearing for each subsystem and numbered with an appropriate subsection designation, such as 9.5.1.1. Under each subsection, further subdivisions may be created at the discretion of the author if needed, particularly if organization by discipline or component is useful for the clarity of the system discussion. Combine the discussion of potential releases from identified subsystems into one section, Section 9.5.3.

9.5.1 Design Bases

10 CFR 60.130 requires that an application to receive, possess, store, and dispose of high-level radioactive waste in the geologic repository operations area must include the principal design criteria for a proposed facility. 10 CFR 60.21(c)(2) requires that the Safety Analysis Report include a description of the design criteria and their relationship to general performance objectives of NRC, design bases and their relation to the principal design criteria, information relative to materials of construction, and codes and standards that will be applied to the geologic repository operations area. Describe the design bases for the mixed low-level radioactive waste management system/subsystems. Include the bases for the systems installed to manage the solid and liquid mixed waste that is hazardous radioactive materials generated at the geologic repository operations area during normal operations, anticipated operational occurrences, and accident situations. Identify the relationship of the bases or criteria to geologic repository operations area performance objectives.

In the design bases description, include information that satisfies each of the following references to the CFR. See Section III of this chapter for the full text of the CFR requirement.

- 10 CFR 60.111(a) requires that the geologic repository operations area be designed so that, until permanent closure, radiation exposures, radiation levels, and releases of radioactive materials to unrestricted areas will be maintained within the limits specified in 10 CFR 20 and such generally applicable environmental standards for radioactivity as may have been established by EPA.

- 10 CFR 60.131(a)(1), (2), (3), (4), and (6) require that the geologic repository operations area be designed to maintain doses, levels, and concentrations of radioactive material in air in restricted areas within the limits specified in 10 CFR 20, and address specific requirements.

- 10 CFR 60.131(b) requires that structures, systems, and components important to safety be designed so that they will perform necessary safety functions, assuming occurrence of design basis events.

- 10 CFR 60.131(c) requires that structures, systems, and components important to safety be designed to withstand dynamic effects, such as missile impacts, that could result from equipment failure, and similar events and conditions that could lead to loss of safety functions.

- 10 CFR 60.131(d)(1), (2), and (4) require implementation of specified fire and explosion protection considerations for structures, systems, and components important to safety.

- 10 CFR 60.131(e)(1) requires that structures, systems, and components important to safety be designed to maintain control of radioactive waste and radioactive effluents and to permit prompt termination of operations during an emergency.

- 10 CFR 60.131(f)(3) requires that provisions be made that, on loss of primary electric power source or circuit, reliable and timely emergency power can be provided to instruments and operating systems, including alarm systems, important to safety.

- 10 CFR 60.131(g) requires that structures, systems, and components important to safety be designed to permit periodic inspection, testing, and maintenance as necessary to ensure their continued functioning and readiness.

- 10 CFR 60.131(i) requires that the design include provisions for instrumentation and control systems to monitor and control the behavior of systems important to safety, assuming occurrence of design basis events.

- 10 CFR 60.132(c)(1) requires that the surface facilities be designed to control the release of radioactive materials in effluents during Category 1 design basis events so as to meet the performance objectives of Section 60.111(a).

- 10 CFR 60.132(d) requires that radioactive waste treatment facilities be designed to process any radioactive wastes generated at the geologic repository operations area into a form suitable to permit safe disposal at the geologic repository operations area or to permit safe transportation and conversion to a form suitable for disposal at an alternative site in accordance with any regulations that are applicable.

- 10 CFR 60.132(e) requires that the surface facility be designed to facilitate decontamination or dismantlement to the same extent as would be required under other parts of that chapter, with respect to equivalent activities licensed thereunder.

- 10 CFR 60.136(b) requires that, for Category 2 design basis events, no individual located on or beyond any point on the boundary of the preclosure controlled area will receive the more limiting of a total effective dose equivalent of 0.05 Sv (5 rem), or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.5 Sv (50 rem). The eye dose equivalent shall not exceed 0.15 Sv (15 rem), and the shallow dose equivalent to skin shall not exceed 0.5 Sv (50 rem).

9.5.2 System Description

Introduce this section with a brief description of the mixed low-level radioactive waste management philosophy. Provide a general design description, referring to site and location/building maps to show how the system relates to and works with other systems.

Provide design criteria for the mixed low-level radioactive waste management system. Obtain these criteria from project documents, such as system description documents.

General Design Criteria—Describe the general design criteria that govern the design of the mixed low-level radioactive waste management system using the technical criteria from Subpart E to 10 CFR 60 and using a similar design philosophy to that contained in the reactor general design criteria of 10 CFR 50, Appendix A. Such criteria should relate to topics such as those that follow:

- **Quality Standards and Records**—State (or refer to another location in this chapter that identifies) the codes and standards to which the system will be designed, fabricated, inspected, and tested to help ensure integrity and high quality; and provide a brief justification for those chosen. Identify which system components will be controlled in accordance with the requirements of DOE/RW-0333P, *Quality Assurance Requirements and Description* (DOE 1998). State that records will be kept for design, construction, and testing of the system and its components. Provide a cross-reference to the appropriate procedure or process that will be used to ensure these records are made and retained. Refer the reader to Chapter 14 of the License Application for additional information on quality assurance.

- **Design Bases for Protection Against Natural Phenomena**—State that the system is designed and will be constructed to withstand the effects of earthquakes and other design natural phenomena (list them) without loss of capability to perform its safety functions. Describe briefly the major design features that provide assurance of protection.

- **Fire Protection**—State that the system is designed and will be constructed to minimize the probability and effect of fires and explosions. Briefly describe the design features that provide this assurance.

- **Environmental and Dynamic Effects Design Basis**—State that the system is designed and will be constructed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. Describe briefly the design features that provide this assurance. Mention the dynamic effects (such as missiles, pipe whipping, and discharging fluids) against which the system is protected. Refer to any information elsewhere in the License Application that supports this discussion.

- **Control of Releases of Radioactive Materials to the Environment**—State that the system is designed to control the release of mixed radioactive material. Describe the means to control the release of effluents produced during normal operation, including anticipated operational occurrences. Include a description of holdup capacity that is provided for retention of mixed waste containing radioactive materials.

- **Radioactivity Control**—State that the system is designed to ensure adequate safety under normal and postulated accident conditions. Briefly describe the design provisions for periodic inspection and testing of components important to safety and for suitable shielding for radiation protection.

For the system or each of the subsystems, include the following information (tabulated information is encouraged where appropriate):

- Purpose of the system.

- A description of the design and performance requirements for structures, systems, and components of the geologic repository that are important to safety (10 CFR 60.21(c)(3)) and a statement that the system meets these performance requirements. Refer to Chapter 7 for detailed analysis and demonstration that the design and performance requirements for Category I and II design basis events have been met.

- Safety classification of the system/subsystem and major components and the rationale for that classification. Refer the License Application reader to Chapter 2 for detailed discussion of the classification process. For major systems/components, include analysis of system/component performance when identifying those important to safety. For the purposes of this analysis, assume that operations at the geologic repository operations area will be carried out at the maximum capacity and rate of receipt of radioactive waste (10 CFR 60.21(c)(1)(ii)(E)). For mixed low-level radioactive waste management system components, list and describe key design parameters, such as duty, capacity, horsepower, safety features, flow, temperature, pressure, and materials of construction.

- Each mixed low-level radioactive waste source and the volume of mixed waste anticipated from each source.

- For each system/subsystem, a description of how the system/subsystem achieves its functions and design bases, flow paths, major components, materials, capacities, collection, temporary storage capabilities, including storage method, capacities, and related facilities, and system/subsystem operations and protective actions, including those intended to prevent release of liquid waste to unrestricted areas.

- Method to be used for processing or treating each mixed waste type, the type of container to be used for packaging, the form of the final processed mixed waste, and the method for disposal. If the methods of disposal include shipment to a burial facility, make a statement that the facility will have the licenses or authorizations required by the governing authority for receipt of the shipped mixed waste.

- The design features and procedures that minimize generation of mixed liquid waste, that minimize the possibility of spills, that provide for control and containment of spills, and that minimize the possibility of spread of contamination.

- The system/subsystem response to normal operations, anticipated operational occurrences, and accident situations.

- Electrical power requirements, including redundancy.

- Instrumentation and control systems associated with the system/subsystem.

- Interfaces with other systems/subsystems.
- Applicable codes and standards.
- Design alternatives that were considered and the rationale for selecting the current design (as applicable).
- ALARA considerations incorporated into the design, including cost/benefit analyses that were performed.
- Relationship of the system to the overall repository performance or to the related NRC regulations.
- A description of system design considerations, if any, that are intended to facilitate system or facility decontamination or dismantlement (10 CFR 60.21(c)(11)).

Provide system diagrams to support and further explain the system description(s). On the drawings, identify and show any radiological design features, such as shielding and thicknesses.

Briefly compare and contrast the mixed low-level radioactive waste management system with the provisions for processing non-radioactive hazardous waste. The purpose of this discussion is to identify that there will be provisions for handling hazardous waste at the geologic repository operations area and that there are similarities and differences with how this waste is handled compared with mixed waste. In general terms, give examples of the types of non-radioactive hazardous waste anticipated and the general approach to processing and disposing of such waste. Do not provide detailed descriptions of systems or handling practices for non-radioactive hazardous waste.

Regulatory Guide 1.143, *Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants* (NRC 1979a), furnishes design guidance acceptable to NRC relating to seismic and quality group classification and quality assurance provisions for radioactive waste management structures, systems, and components. It also describes provisions for controlling releases of liquids containing radioactive materials, such as spills or tank overflows, from all systems. Regulatory Position 1 of this guide addresses systems handling radioactive materials in liquids; Regulatory Position 3 addresses solid radioactive waste systems; Regulatory Position 4 identifies additional design, construction, and testing criteria; Regulatory Position 5 addresses seismic design for radioactive waste management systems and structures housing radioactive waste management systems; and Regulatory Position 6 identifies quality assurance for radioactive waste management systems. Review this guide. Some of the guidance is potentially useful and applicable. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of this technical guidance document.

10 CFR 20.1101(b) requires that the licensee use, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are ALARA. Regulatory Guide 8.8, *Information*

Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable (NRC 1978f), provides information relevant to attaining ALARA goals and objectives. Regulatory Position 2, Facility and Equipment Design Features, of that regulatory guide contains practices for incorporating ALARA principles into the design of facilities. State the extent (if any) that the geologic repository operations area design is consistent/compliant with these aspects of regulatory guidance. Describe procedures, engineering controls, and specific design features that apply to the mixed low-level radioactive waste management system/subsystems that will be incorporated to apply ALARA principles to those systems. Relate specific design features to those discussed in Regulatory Position 2 of Regulatory Guide 8.8, as applicable. Identify ALARA features that apply to dose reduction, both of the public and of the occupational worker.

If the system/subsystem design includes concrete radiation shields, Regulatory Guide 1.69, *Concrete Radiation Shields for Nuclear Power Plants* (NRC 1973a), which implements ANSI N101.6-1972, *Concrete Radiation Shields* (ANSI 1972), with specified exceptions, may provide useful and applicable guidance. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of the technical guidance document.

Cost-benefit analyses are considered by NRC to be an integral part of a viable ALARA program. Lack of cost-benefit analyses related to ALARA design features is considered to be de facto evidence that the ALARA program is not properly implemented. Describe in this section any cost-benefit analyses associated with the design of the liquid low-level radioactive waste system/subsystems. Regulatory Guide 1.110, *Cost-Benefit Analysis for Radioactive Waste Systems for Light-Water-Cooled Nuclear Power Reactors* (NRC 1976c), provides guidance on acceptable practices for cost-benefit analyses that involve consideration of reduction in dose to the general public. Review this guide and standard. Some of the guidance is potentially useful and applicable. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of this technical guidance document.

9.5.3 Design Evaluation

For each source identified in Section 9.1 for the mixed low-level radioactive waste management system(s), identify the expected releases of mixed waste, the bases for the values used, and release paths during normal operations, anticipated operational occurrences, and accident situations.[7] Identify the parameters and assumptions used for the release calculations. Tabulate the releases by radionuclide for each subsystem identified above and for the total system, and indicate effluent concentrations. With references to Chapter 7, compare the doses due to the effluents with the numerical design objectives of 10 CFR 50, Appendix I, and to the system design bases (specifically to those associated with dose limits of 10 CFR 20 and to EPA

[7] Dose assessments to the public associated with releases to the environment are provided in Chapter 7 and should be referenced in this chapter as appropriate. Dose assessments to occupational workers based on gaseous/particulate effluent releases are provided in Chapter 10.

preclosure dose limits (if they are promulgated)) (see 10 CFR 60.111). This dose comparison should address dose contribution from mixed waste as well as a total dose from all sources. If no mixed radioactive releases are anticipated under any or all of these operational conditions, then make a statement to that effect and explain.

9.6 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

The process and effluent radiological monitoring and sampling systems may be addressed as a whole or may be divided into subsystems for discussion of each topic below. If the system is divided into subsystems, organize this section so that each of the subsystems is identified under a subsection, such as 9.6.1, with each of the following headings (design bases, system description, and design evaluation) appearing for each subsystem and numbered with an appropriate subsection designation, such as 9.6.1.1. Under each subsection, further subdivisions may be created at the discretion of the author if needed, particularly if organization by discipline or component is useful for the clarity of the system discussion. Combine the discussion of potential releases from identified subsystems into one section, Section 9.6.3.

9.6.1 Design Bases

10 CFR 60.130 requires that an application to receive, possess, store, and dispose of high-level radioactive waste in the geologic repository operations area must include the principal design criteria for a proposed facility. 10 CFR 60.21(c)(2) requires that the Safety Analysis Report include a description of the design criteria and their relationship to general performance objectives of NRC, design bases and the relation of them to the principal design criteria, information relative to materials of construction, and codes and standards that will be applied to the geologic repository operations area. Describe the design bases for the process and effluent radiological monitoring and sampling systems. Include the bases for the systems that monitor and sample potentially radioactive gases and air particulates from process vents and heating, ventilation, and air-conditioning (HVAC) effluent stacks and that monitor process liquids in liquid low-level radioactive waste management systems during normal operations, anticipated operational occurrences, and accident situations. Identify the relationship of the bases to geologic repository operations area performance objectives. Make a clear statement that no liquid effluents are expected, so liquid effluent monitoring systems are not planned. Refer to Section 10.2.4 for a description of the area radiation and airborne radioactivity monitoring instrumentation and the common systems that respond if radiation is detected.

The process and effluent radiological monitoring and sampling systems provide monitoring to facilitate, monitor, and verify conformance with limits to dose to members of the public that are specified in the CFR. Include the design bases of this system that contribute to meeting the following CFR requirements.

- 10 CFR 20.1301 specifies dose limits to members of the public that result from geologic repository operations area operations.

- 10 CFR 20.1101(d) requires that the licensee, as part of the requirements for implementing Section 2011.01(b), constrain air emissions to limit doses from those

emissions. These constraints apply in addition to the requirements of Section 20.1301.

- 10 CFR 20.1302(a) requires that the licensee make surveys of radioactive materials in effluents released to unrestricted and controlled areas to demonstrate compliance with dose limits for members of the public in Section 20.1301. 10 CFR 20.1302(b) requires that the licensee show compliance with the limits of Section 20.1301 by measurement or calculation that doses and concentrations do not exceed specified limits. 10 CFR 20.1302(c) allows, with NRC approval, adjustment of effluent concentration values to take into account actual characteristics of the effluents.

- 10 CFR 20.1501(b) requires that the licensee ensure that instruments and equipment used for quantitative radiation measurements, such as effluent monitoring, are calibrated periodically for the radiation measured.

- 10 CFR 60.111(a) requires that the geologic repository operations area be designed so that, until permanent closure, radiation exposures, radiation levels, and releases of radioactive materials to unrestricted areas will be maintained within the limits specified in 10 CFR 20 and such generally applicable environmental standards for radioactivity as may have been established by EPA.

- 10 CFR 60.131(a)(1), (2), (3), (4), and (6) require that the geologic repository operations area be designed to maintain doses, levels, and concentrations of radioactive material in air in restricted areas within the limits specified in 10 CFR 20, and address specific requirements.

- 10 CFR 60.131(b) requires that structures, systems, and components important to safety be designed so that they will perform necessary safety functions, assuming occurrence of design basis events.

- 10 CFR 60.131(c) requires that structures, systems, and components important to safety be designed to withstand dynamic effects, such as missile impacts, that could result from equipment failure, and similar events and conditions that could lead to loss of safety functions.

- 10 CFR 60.131(d)(1), (2), and (4) require implementation of specified fire and explosion protection considerations for structures, systems, and components important to safety.

- 10 CFR 60.131(e)(1) requires that structures, systems, and components important to safety be designed to maintain control of radioactive waste and radioactive effluents and to permit prompt termination of operations during an emergency.

- 10 CFR 60.131(f)(3) requires that provisions be made that, on loss of primary electric power source or circuit, reliable and timely emergency power can be provided to instruments and operating systems, including alarm systems, important to safety.

- 10 CFR 60.131(g) requires that structures, systems, and components important to safety be designed to permit periodic inspection, testing, and maintenance as necessary to ensure their continued functioning and readiness.

- 10 CFR 60.131(i) requires that the design include provisions for instrumentation and control systems to monitor and control the behavior of systems important to safety, assuming occurrence of design basis events.

- 10 CFR 60.132(c)(2) requires that the effluent monitoring systems be designed to measure the amount and concentration of radionuclides in any effluent with sufficient precision to determine whether releases conform to the design requirement for effluent control. The monitoring systems shall be designed to include alarms that can be periodically tested.

- 10 CFR 60.132(e) requires that the surface facility be designed to facilitate decontamination or dismantlement to the same extent as would be required, under other parts of that chapter, with respect to equivalent activities licensed thereunder.

- 10 CFR 60.136(b) requires that, for Category 2 design basis events, no individual located on or beyond any point on the boundary of the preclosure controlled area will receive the more limiting of a total effective dose equivalent of 0.05 Sv (5 rem), or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.5 Sv (50 rem). The eye dose equivalent shall not exceed 0.15 Sv (15 rem), and the shallow dose equivalent to skin shall not exceed 0.5 Sv (50 rem).

9.6.2 System Description

This section will provide descriptions of the systems used to monitor and sample radiological process and effluent gases and of process liquids.

Provide design criteria for the systems used to monitor and sample radiological process and effluent gases and of process liquids. Obtain these criteria from project documents, such as system description documents.

General Design Criteria—Describe the general design criteria that govern the design of the process and effluent radiological monitoring and sampling system using the technical criteria from 10 CFR 60, Subpart E, and using a similar design philosophy to that contained in the reactor general design criteria of 10 CFR 50, Appendix A. Such criteria should relate to topics such as those that follow:

- **Quality Standards and Records**—State (or refer to another location in this chapter that identifies) the codes and standards to which the system will be designed, fabricated, inspected, and tested to help ensure integrity and high quality; and provide a brief justification for those chosen. Identify which system components will be controlled in accordance with the requirements of DOE/RW-0333P, *Quality Assurance Requirements and Description* (DOE 1998). State that records will be

kept for design, construction, and testing of the system and its components. Provide a cross-reference to the appropriate procedure or process that will be used to ensure these records are made and retained. Refer the reader to Chapter 14 of the License Application for additional information on quality assurance.

- **Design Bases for Protection Against Natural Phenomena**—State that the system is designed and will be constructed to withstand the effects of earthquakes and other design natural phenomena (list them) without loss of capability to perform its safety functions. Describe briefly the major design features that provide assurance of protection.

- **Fire Protection**—State that the system is designed and will be constructed to minimize the probability and effect of fires and explosions. Briefly describe the design features that provide this assurance.

- **Environmental and Dynamic Effects Design Basis**—State that the system is designed and will be constructed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. Describe briefly the design features that provide this assurance. Mention the dynamic effects (such as missiles, pipe whipping, and discharging fluids) against which the system is protected. Refer to any information elsewhere in the License Application that supports this discussion.

- **Control of Releases of Radioactive Materials to the Environment**—State that the system is designed to monitor and control the release of radioactive material. Describe the features that ensure this control during normal operation, including anticipated operational occurrences.

- **Radioactivity Control**—State that the system is designed to ensure adequate safety under normal and postulated accident conditions. Briefly describe the design provisions for periodic inspection and testing of components important to safety and for suitable shielding for radiation protection.

- **Monitoring Radioactivity Releases**—Describe the means provided for monitoring effluent discharge paths for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

For the system or each of the subsystems, include the following information (tabulated information is encouraged where appropriate):

- Purpose of the system/major subsystems.

- A description of the design and performance requirements for structures, systems, and components of the geologic repository that are important to safety (10 CFR 60.21(c)(3)) and a statement that the system meets these performance requirements.

Refer to Chapter 7 for detailed analysis and demonstration that the design and performance requirements for Category I and II design basis events have been met.

- Safety classification of the system/subsystem and major components and the rationale for that classification. Refer the License Application reader to Chapter 2 for detailed discussion of the classification process. For major systems/components, include analysis of system/component performance when identifying those important to safety. For the purposes of this analysis, assume that operations at the geologic repository operations area will be carried out at the maximum capacity and rate of receipt of radioactive waste (10 CFR 60.21(c)(1)(ii)(E)). For process and effluent radiological monitoring and sampling system components, list and describe key design parameters, such as duty, capacity, horsepower, safety features, flow, temperature, pressure, and materials of construction.

- For each system/subsystem, a description of how the system/subsystem achieves its functions and meets design bases.

- Instrumentation, redundancy, independence, and diversity of the components.

- Range of radioactivity concentrations to be monitored and bases for ranges identified.

- Locations and types of annunciators and alarms. If automatic controls and actions initiated by alarms are known, describe them (required for the application to receive and possess high-level radioactive waste).

- Provisions for emergency power.

- General information on planned provisions for instrument calibration, maintenance, inspection, decontamination, and replacement.

- ALARA considerations incorporated into the design.

- A description of how representative samples from effluent monitors will be obtained.

- A description of system design considerations, if any, that are intended to facilitate system or facility decontamination or dismantlement (10 CFR 60.21(c)(11)).

Provide system diagrams to support the system description(s) and show the location of radiological monitoring instrumentation. Relate these systems to the flow diagrams for the gaseous and liquid low-level radioactive waste management systems. On the drawings, identify and show any radiological design features, such as shielding and thicknesses.

10 CFR 20.1101(b) requires that the licensee use, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are ALARA. Regulatory Guide 8.8, *Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable* (NRC 1978f), provides information relevant to attaining

ALARA goals and objectives. Regulatory Position 2, Facility and Equipment Design Features, of that regulatory guide contains practices for incorporating ALARA principles into the design of facilities. Make a statement that indicates the extent (if any) that the geologic repository operations area design is consistent/compliant with these aspects of regulatory guidance. Describe procedures, engineering controls, and specific design features that apply to the process and effluent radiological monitoring and sampling systems that will be incorporated to apply ALARA principles to those systems. Relate specific design features to those discussed in Regulatory Position 2 of Regulatory Guide 8.8 (NRC 1978f), as applicable. Identify ALARA features that apply to dose reduction both of the public and of the occupational worker.

Regulatory Guide 8.37, *ALARA Levels for Effluents from Materials Facilities* (NRC 1993), Regulatory Position 2, Procedures, Engineering Controls, and Process Controls, addresses using available engineering options to control the releases of effluents to the environment. Regulatory Position 3, Surveys and Effluent Monitoring, of this regulatory guide states that effluent monitoring systems should be designed in accordance with ANSI N13.1-1969, *Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities* (ANSI 1969) and ANSI N42.18, *Specification and Performance of Onsite Instrumentation for Continuously Monitoring Radioactive Effluents* (ANSI 1980). Review this guide and its implemented standards; some of its guidance is potentially useful and applicable. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of this technical guidance document.

Regulatory Guide 1.97, *Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident* (NRC 1983), provides methodology acceptable to NRC for providing instrumentation to monitor variables and systems during and following an accident. Much of this guidance is associated with catastrophic accidents at nuclear power plants and the instrumentation requirements for the control room operators to anticipate these accidents and mitigate them. However, Regulatory Position 2, Instrumentation for Monitoring Systems Operation and Effluent Release, of this guide provides guidance on instrumentation to determine the magnitude of releases of radioactive materials and to continuously assess such releases. Review this guide. Some of the guidance is potentially useful and applicable. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of this technical guidance document.

Regulatory Guide 1.143, *Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants* (NRC 1979a), furnishes design guidance acceptable to NRC relating to seismic and quality group classification and quality assurance provisions for radioactive waste management structures, systems, and components. It also describes provisions for controlling releases of liquids containing radioactive materials, such as spills or tank overflows, from all systems. Regulatory Position 1 of this guide addresses requirements for liquid radioactive waste systems, Regulatory Position 2 of this guide addresses systems handling radioactive materials in gases; Regulatory Position 4 identifies additional design, construction, and testing criteria; Regulatory Position 5 addresses seismic design for radioactive waste management systems and structures housing radioactive waste management systems; and Regulatory Position 6 identifies quality assurance

for radioactive waste management systems. Review this guide. Some of the guidance is potentially useful and applicable. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of this technical guidance document.

Regulatory Guide 4.20, *Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees Other Than Power Reactors* (NRC 1996a), provides guidance on methods acceptable to NRC for compliance with the combination of the requirements in 10 CFR 20.1302 (to demonstrate compliance with annual dose limits) and 10 CFR 20.1101(d) (to implement ALARA principles with a further constraint on air emissions). Regulatory Position 2 of this guide provides methodology and information for screening technique and demonstration of compliance with the constraint on air emissions. Review this guide. Some of the guidance is potentially useful and applicable. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of this technical guidance document.

If the system/subsystem design includes concrete radiation shields, include a statement that the concrete shields conform to the requirements of Regulatory Guide 1.69, *Concrete Radiation Shields for Nuclear Power Plants* (NRC 1973a), which implements ANSI N101.6-1972, *Concrete Radiation Shields* (ANSI 1972), with specified exceptions. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of the technical guidance document.

Cost-benefit analyses are considered by NRC to be an integral part of a viable ALARA program. Lack of cost-benefit analyses related to ALARA design features is considered to be de facto evidence that the ALARA program is not properly implemented. Describe in this section any cost-benefit analyses associated with the design of the process and effluent radiological monitoring and sampling systems.

Regulatory Guide 1.21, *Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants* (NRC 1974a), describes programs acceptable to NRC for measuring, reporting, and evaluating releases of radioactive materials in liquid and gaseous effluents and guidelines for classifying and reporting the categories and curie content of solid low-level radioactive wastes. It includes applicable guidance on the location of monitoring, the type of monitoring, acceptability of gross versus specific radioactivity measurements, and representative samples. It also addresses effluent and waste disposal reporting. Review this guide. Some of the guidance is potentially useful and applicable. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of this technical guidance document.

9.6.3 Effluent Monitoring and Sampling Operations

10 CFR 60.21(c)(7) requires that the Safety Analysis Report include a description of the program for control and monitoring of radioactive effluents to maintain such effluents in accordance with the 10 CFR 20 requirements. Describe the program and the analytical approaches to be taken to monitor the radioactive material content of the effluent streams of the geologic repository

operations area facilities. Relate the monitoring program to process flow diagrams. Describe how effluents will be maintained within limits.

For gaseous and particulate effluents, describe the features of the monitoring systems to be used; the release paths to be monitored; the expected locations of the monitors; the types of monitors, measurements made, such as gross alpha or isotopic analysis; and the approaches used to assess the radioactive material content. Discuss the reliability and sensitivity of each measurement to be made, and justify the selection of each system/location sampled and instrument used for measurement. Discuss the frequency of sampling, the limits for action, and the plans to be used to maintain continued integrity of analyses. Tabulate information as needed for clarity.

For liquid effluents, describe the features of the monitoring systems to be used, their locations, and the items to be monitored. For each system, show the expected reliability and sensitivity. Justify the selection of each system/location sampled and instrument used for measurement. Discuss the frequency of sampling, the limits for action, and the plans to be used to maintain continued integrity of analyses.

For solid and mixed waste, describe the procedures, equipment, and instrumentation used to monitor all solid and mixed radioactive waste.

Regulatory Guide 8.37, *ALARA Levels for Effluents from Materials Facilities* (NRC 1993), Regulatory Position 1, ALARA Goals, addresses a method acceptable to NRC to demonstrate compliance in establishing ALARA goals for effluents. Regulatory Position 2 addresses procedures, engineering controls, and process controls to control the release of effluents to the environment. Regulatory Position 3 addresses surveys and effluent monitoring. Review this guide. Some of the guidance is potentially useful and applicable. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of this technical guidance document.

Working with the author of Chapter 10 of the License Application, address the program to ensure the quality of the results of measurements of radioactive materials in effluents. Regulatory Guide 4.15, *Quality Assurance for Radiological Monitoring Programs (Normal Operations) Effluent Streams and the Environment* (NRC 1979b), provides guidance for establishing such a program. Some of the positions addressed are appropriately addressed in Chapter 10 of the License Application. Regulatory Positions 6, Quality Control in the Radio Analytical Laboratory, and 7, Quality Control for Continuous Effluent Monitoring Systems, are potentially applicable to this chapter. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of the technical guidance document.

9.6.4 Design Evaluation

Describe the estimated contribution of radioactive materials to the environment from the geologic repository operations area and the methods used to make the estimate. Identify the methodology to be used for calculating the releases of radioactive materials in gaseous and liquid effluents and the parameters used to calculate the quantities of radioactive materials in gaseous and liquid effluents from the geologic repository operations area. Regulatory Guide 1.112,

Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors (NRC 1976a), provides references to two NUREGs[8] that provide acceptable methods for calculating annual average expected releases of radioactive material in liquid and gaseous effluents. Review this guide and the referenced NUREGs. Some of the guidance is potentially useful and applicable. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of this technical guidance document.

V. REFERENCES

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

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

10 CFR 60. Energy: Disposal of High-Level Radioactive Wastes in Geologic Repositories.

ANSI (American National Standards Institute) 1969. *Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities.* ANSI N13.1-1969: La Grange Park, Illinois. 204994.

ANSI 1972. *Concrete Radiation Shields.* ANSI N101.6-1972. La Grange Park, Illinois. 233869.

ANSI 1975. *Supplementary Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Power Plants.* ANSI N45.2.8-1975. La Grange Park, Illinois. 236056.

ANSI 1980. *Specification and Performance of Onsite Instrumentation for Continuously Monitoring Radioactive Effluents.* ANSI N42.18. La Grange Park, Illinois. 205004.

CRWMS M&O (Civilian Radioactive Waste Management System Management and Operating Contractor) 1996. *Mined Geologic Disposal System Advanced Conceptual Design Report.* B00000000-01717-5705-00027. Rev. 0. Las Vegas, Nevada. 225721.

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

NRC (U.S. Nuclear Regulatory Commission) 1972. *Control of Electroslag Weld Properties.* Regulatory Guide 1.34. Rev. 0. Washington, D.C.: U.S. Government Printing Office. 4645.

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[8] The two NUREGs identified are NUREG-0016, *Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWR-GALE Code)* (NRC 1978a); and NUREG-0017, *Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code)* (NRC 1985).

NRC 1973a. *Concrete Radiation Shields for Nuclear Power Plants.* Regulatory Guide 1.69. Rev. 0. Washington, D.C.: U.S. Government Printing Office. 222690.

NRC 1973b. *Control of Preheat Temperature for Welding of Low-Alloy Steel.* Regulatory Guide 1.50. Rev. 0. Washington, D.C.: U.S. Government Printing Office. 4547.

NRC 1973c. *Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components.* Regulatory Guide 1.43. Rev. 0. Washington, D.C.: U.S. Government Printing Office. 4551.

NRC 1973d. *Control of the Use of Sensitized Stainless Steel.* Regulatory Guide 1.44. Rev. 0. Washington, D.C.: U.S. Government Printing Office. 4549.

NRC 1973e. *Damping Values for Seismic Design of Nuclear Power Plants.* Regulatory Guide 1.61. Rev. 0. Washington, D.C.: U.S. Government Printing Office. 2710.

NRC 1973f. *Design Response Spectra for Seismic Design of Nuclear Power Plants.* Regulatory Guide 1.60. Rev. 1. Washington, D.C.: U.S. Government Printing Office. 232770.

NRC 1973g. *Nonmetallic Thermal Insulation for Austenitic Stainless Steel.* Regulatory Guide 1.36. Rev. 0. Washington, D.C.: U.S. Government Printing Office. 4646.

NRC 1973h. *Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants.* Regulatory Guide 1.54. Rev. 0. Washington, D.C.: U.S. Government Printing Office. 4546.

NRC 1974a. *Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants.* Regulatory Guide 1.21. Rev. 1. Washington, D.C.: U.S. Government Printing Office. 223043.

NRC 1974b. *Process Offgas Systems for Fuel Reprocessing Plants.* Regulatory Guide 3.20. Rev. 0. Washington, D.C.: U.S. Government Printing Office. 4570.

NRC 1975a. *Design Considerations for Minimizing Residual Holdup of Special Nuclear Material in Equipment for Dry Process Operations.* Regulatory Guide 5.42. Rev. 0. Washington, D.C.: U.S. Government Printing Office. 4707.

NRC 1975b. *Guidance for Avoiding Intergranular Corrosion and Stress Corrosion in Austenitic Stainless Steel Components of Fuel Reprocessing Plants.* Regulatory Guide 3.37. Rev. 0. Washington, D.C.: U.S. Government Printing Office. 4579.

NRC 1975c. *Selection, Application, and Inspection of Protective Coatings (Paints) for Fuel Reprocessing Plants.* Regulatory Guide 3.30. Rev. 0-R. Washington, D.C.: U.S. Government Printing Office. 4587.

NRC 1976a. *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. Government Printing Office. 222664.

NRC 1976b. *Combining Modal Responses and Spatial Components in Seismic Response Analysis*. Regulatory Guide 1.92. Rev. 1. Washington, D.C.: U.S. Government Printing Office. 2775.

NRC 1976c. *Cost-Benefit Analysis for Radioactive Waste Systems for Light-Water-Cooled Nuclear Power Reactors*. Regulatory Guide 1.110. Rev. 0. Washington, D.C.: U.S. Government Printing Office. 2779.

NRC 1976d. *Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment*. Regulatory Guide 1.116. Rev. 0-R. Washington, D.C.: U.S. Government Printing Office. 4632.

NRC 1977a. *Control of Ferrite Content in Stainless Steel Weld Metal*. Regulatory Guide 1.31. Rev 3. Washington, D.C.: U.S. Government Printing Office. 4643.

NRC 1977b. *Housekeeping Requirements for Water-Cooled Nuclear Power Plants*. Regulatory Guide 1.39. Rev. 2. Washington, D.C.: U.S. Government Printing Office. 4641.

NRC 1977c. *Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants*. Regulatory Guide 1.38. Rev. 2. Washington, D.C.: U.S. Government Printing Office. 4648.

NRC 1978a. *Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWR-GALE Code)*. NUREG-0016. Washington, D.C.: U.S. Government Printing Office. 210607.

NRC 1978b. *Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components*. Regulatory Guide 1.122. Rev. 1. Washington, D.C.: U.S. Government Printing Office. 2787.

NRC 1978c. *Seismic Design Classification*. Regulatory Guide 1.29. Rev. 3. Washington, D.C.: U.S. Government Printing Office. 226259.

NRC 1978d. *Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports*. Regulatory Guide 1.124. Rev. 1. Washington, D.C.: U.S. Government Printing Office. 2939.

NRC 1978e. *Service Limits and Loading Combinations for Class 1 Plate-And-Shell-Type Component Supports*. Regulatory Guide 1.130. Rev. 1. Washington, D.C.: U.S. Government Printing Office. 4598.

NRC 1978f. *Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable*. Regulatory Guide 8.8. Rev. 3. Washington, D.C.: U.S. Government Printing Office. 2887.

NRC 1979a. *Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants*. Regulatory Guide 1.143. Rev. 1. Washington, D.C.: U.S. Government Printing Office. 2790.

NRC 1979b. *Quality Assurance for Radiological Monitoring Programs (Normal Operations) Effluent Streams and the Environment.* Regulatory Guide 4.15. Rev. 1. Washington, D.C.: U.S. Government Printing Office. 2951.

NRC 1979c. *Site Investigations for Foundations of Nuclear Power Plants.* Regulatory Guide 1.132. Rev. 1. Washington, D.C.: U.S. Government Printing Office. 218521.

NRC 1981a. *Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants.* Regulatory Guide 1.148. Rev. 0. Washington, D.C.: U.S. Government Printing Office. 4668.

NRC 1981b. *Materials, Construction, and Testing of Concrete Containments.* Regulatory Guide 1.136. Rev. 2. Washington, D.C.: U.S. Government Printing Office. 4543.

NRC 1983. *Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.* Regulatory Guide 1.97. Rev. 3. Washington, D.C.: U.S. Government Printing Office. 2777.

NRC 1985. *Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code).* NUREG-0017. Washington, D.C.: U.S. Government Printing Office. 210608.

NRC 1988. *Seismic Qualification of Electrical and Mechanical Equipment for Nuclear Power Plants.* Regulatory Guide 1.100. Rev. 2. Washington, D.C.: U.S. Government Printing Office. 4636.

NRC 1989. *Standard Format and Content for a Topical Safety Report for a Spent Fuel Dry Storage Cask.* Regulatory Guide 3.61. Rev. 0. Washington, D.C.: U.S. Government Printing Office. 1368.

NRC 1990. *Format and Content for the License Application for the High-Level Waste Repository.* Regulatory Guide DG-3003. Washington, D.C.: U.S. Government Printing Office. 223107.

NRC 1993. *ALARA Levels for Effluents from Materials Facilities.* Regulatory Guide 8.37. Rev. 0. Washington, D.C.: U.S. Government Printing Office. 9819.

NRC 1995. *License Application Review Plan for a Geologic Repository for Spent Nuclear Fuel and High-Level Radioactive Waste.* NUREG-1323. Rev. 1. Washington, D.C.: U.S. Government Printing Office. 213043.

NRC 1996a. *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. Government Printing Office. 229689.

NRC 1996b. *Standard Review Plan for Spent Fuel Dry Storage Facilities.* NUREG-1567. Washington, D.C.: U.S. Government Printing Office. 226657.

NRC 1997. *Standard Review Plan for Dry Cask Storage Systems.* NUREG-1536. Washington, D.C.: U.S. Government Printing Office. 232373.

YMP (Yucca Mountain Site Characterization Project) 1996. *Topical Report: Preclosure Seismic Design Methodology for a Geologic Repository at Yucca Mountain.* YMP/TR-003-NP. Rev. 1. Las Vegas, Nevada: Yucca Mountain Site Characterization Office. MOL.19970114.0027.

YMP 1997. *Management Plan for the Development of a Project Integrated Safety Assessment (PISA).* YMP/97-01Q. Rev. 2. Las Vegas, Nevada: Yucca Mountain Site Characterization Office. MOL.19971017.0200.

CHAPTER 10. RADIATION PROTECTION

I. OBJECTIVE

This chapter provides guidance for the author(s) of Chapter 10 of the License Application. The License Application will be submitted to the U.S. Nuclear Regulatory Commission (NRC) for authorization to construct a geologic high-level radioactive waste repository at Yucca Mountain. The author(s) must demonstrate that the U.S. Department of Energy (DOE) has workable concepts for a radiation protection program which, if implemented, will allow storage of the identified high-level radioactive waste without exceeding occupational and public dose limits prescribed in the regulations, while maintaining that occupational dose and dose to the public is as low as is reasonably achievable (ALARA). This chapter also describes the environmental monitoring program to be implemented at the repository. Since concepts and applications of radiation protection and environmental monitoring have substantial licensing precedent, the required level of detail is limited, unless otherwise specified. It is counterproductive to provide more detail in the License Application beyond that needed to support NRC reasonable assurance findings. The author must provide sufficient information to support the licensing/safety case and to show compliance with regulations, and generally nothing more.

II. GUIDANCE TOPICS

The following topics will be discussed in Chapter 10 of the License Application:

- 10.0 PURPOSE, SUMMARY, AND CONCLUSIONS
- 10.1 RADIATION PROTECTION PROGRAMS
  - 10.1.1 Policy Considerations
  - 10.1.2 Operational Health Physics
  - 10.1.3 Low-Level Radioactive Waste Management
- 10.2 AS LOW AS REASONABLY ACHIEVABLE DESIGN FEATURES
  - 10.2.1 Facility Design Features
  - 10.2.2 Shielding
  - 10.2.3 Ventilation
  - 10.2.4 Radiological Monitoring Instrumentation
- 10.3 RADIATION PROTECTION OF THE PUBLIC AND ENVIRONMENT
- 10.4 DOSE ASSESSMENT
  - 10.4.1 Radiation Sources
  - 10.4.2 Dose Assessment

III. REGULATORY REQUIREMENTS

***** IMPORTANT NOTE *****

At the time of development of this document, it appears likely that NRC will issue a new, site-specific regulation for Yucca Mountain in 1999. At the time the new 10 CFR 63 is issued, 10 CFR 60 will be modified to state that it is not applicable to Yucca Mountain. Details about the new regulation are not known. However, it has been determined prudent to develop the working draft License Application in accordance with the best available estimate of the form of the new regulation. To avoid confusing references and to allow production of the technical guidance document in time to support training for the working draft License Application, the text of the technical guidance document that follows this note has not been revised to reflect the postulated form of the new regulation. Instead, this note provides guidance needed by working draft License Application authors to comply with management direction on addressing the new regulation. The paragraphs that follow explain changes in the regulation with respect to the existing 10 CFR 60 that affect this technical guidance document chapter and provide guidance on how to address the changes in the working draft License Application. Where guidance in this note conflicts with guidance in the remainder of this chapter of the technical guidance document, the guidance in the note is to govern. Any questions should be addressed to Management and Operating Contractor (M&O) Licensing.

Changes to Regulations Applicable to Chapter 10

The following 10 CFR 60 requirements applicable to technical guidance document Chapter 10 and listed below in Section III are assumed to not be carried over to the new regulation: 60.131 and 60.133.

In place of these specific design criteria, the new regulation is expected to require a discussion of the contribution of the design of the geologic repository operations area to the overall performance of the repository for Category 1 accidents and Category 2 events. The performance objective for Category 1 accidents is expected to be the 10 CFR 20 radiation exposure limits. For Category 2 events the expected performance objective is for radiation exposures to be below 5 rem total effective dose equivalent. The manner in which the design contribution is to be described has not been specified, though an NRC review plan on the subject is expected to be issued sometime in the future.

Guidance to Working Draft License Application Authors for Addressing Changes

- Disregard guidance in Section IV regarding demonstrating compliance with any of the regulations assumed to be deleted as described above. Instead, discuss the relevance of repository design to radiation protection for workers, the public, and the environment.

- Address the technical issue of each of the regulations in Section III, including those expected to be deleted, but do not relate the discussion to demonstrating compliance with any of the deleted regulations. Continue to explicitly address the regulations in Section III that are not expected to be deleted.

- Discussion of design criteria, design bases, and performance is still required as per the existing guidance in Section IV. The discussions should include sufficient detail to ensure that the expected performance objectives for radiation exposures are achieved.

- No sections or subsections of Chapter 10 are expected to be deleted as a result of this note.

***** END OF NOTE *****

Sections of the regulations that are to be addressed fully or in part in Chapter 10 of the License Application are reproduced below. These are verbatim extracts from the regulations. Paragraph numbers refer to Title 10 of the U.S. Code of Federal Regulations (CFR). In addition, Appendix A provides a cross-reference between the regulations in 10 CFR 60 and the specific License Application sections that address these regulations. Revisions of regulations are not uncommon. Verify that current versions are used.

10 CFR Part 19, Notices, Instructions and Reports To Workers: Inspection and Investigations

19.11 Posting of notices to workers.

(a) Each licensee shall post current copies of the following documents:

(1) The regulations in this part and in part 20 of this chapter;

(2) The license, license conditions, or documents incorporated into a license by reference, and amendments thereto;

(3) The operating procedures applicable to licensed activities;

(4) Any notice of violation involving radiological working conditions, proposed imposition of civil penalty, or order issued pursuant to subpart B of part 2 of this chapter, and any response from the licensee.

(b) If posting of a document specified in paragraph (a) (1), (2) or (3) of this section is not practicable, the licensee may post a notice which describes the document and states where it may be examined.

(c)(1) Each licensee and each applicant for a specific license shall prominently post NRC Form 3 (Revision dated January 1996), "Notice to Employees."

(2) Copies of NRC Form 3 may be obtained by writing to the Regional Administrator of the appropriate U.S. Nuclear Regulatory Commission Regional Office listed in Appendix D to Part 20 of this chapter or by calling the NRC Information and Records Management Branch at (301) 415-7230.

(d) Documents, notices, or forms posted pursuant to this section shall appear in a sufficient number of places to permit individuals engaged in licensed activities to observe them on the way to or from any particular licensed activity location to which the

document applies, shall be conspicuous, and shall be replaced if defaced or altered.

(e) Commission documents posted pursuant to paragraph (a)(4) of this section shall be posted within 2 working days after receipt of the documents from the Commission; the licensee's response, if any, shall be posted within 2 working days after dispatch by the licensee. Such documents shall remain posted for a minimum of 5 working days or until action correcting the violation has been completed, whichever is later.

19.12 Instruction to workers.

(a) All individuals who in the course of employment are likely to receive in a year an occupational dose in excess of 100 mrem (1 mSv) shall be-

(1) Kept informed of the storage, transfer, or use of radiation and/or radioactive material;...

(6) Advised as to the radiation exposure reports which workers may request pursuant to §19.13.(b) In determining those individuals subject to the requirements of paragraph (a) of this section, licensees must take into consideration assigned activities during normal and abnormal situations involving exposure to radiation and/or radioactive material which can reasonably be expected to occur during the life of a licensed facility. The extent of these instructions must be commensurate with potential radiological health protection problems present in the work place.

10 CFR Part 20, Standards For Protection Against Radiation
Subpart B - Radiation Protection Programs
20.1101 Radiation protection programs.

(a) Each licensee shall develop, document, and implement a radiation protection program commensurate with the scope and extent of licensed activities and sufficient to ensure compliance with the provisions of this part. (See §20.2102 for recordkeeping requirements relating to these programs.)

(b) The licensee shall use, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA).

(c) The licensee shall periodically (at least annually) review the radiation protection program content and implementation.

(d) To implement the ALARA requirements of §20.1101(b), and notwithstanding the requirements in §20.1301 of this part, a constraint on air emissions of radioactive material to the environment, excluding Radon-222 and its daughters, shall be established by licensees other than those subject to §50.34a, such that the individual member of the public likely to receive the

highest dose will not be expected to receive a total effective dose equivalent in excess of 10 mrem (0.1 mSv) per year from these emissions. If a licensee subject to this requirement exceeds this dose constraint, the licensee shall report the exceedance as provided in §20.2203 and promptly take appropriate corrective action to ensure against recurrence.

Subpart C - Occupational Dose Limits

20.1201 Occupational dose limits for adults.

(a) The licensee shall control the occupational dose to individual adults, except for planned special exposures under §20.1206, to the following dose limits.

(1) An annual limit, which is the more limiting of-

(i) The total effective dose equivalent being equal to 5 rems (0.05 Sv); or

(ii) The sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye being equal to 50 rems (0.5 Sv).

(2) The annual limits to the lens of the eye, to the skin, and to the extremities, which are:

(i) An eye dose equivalent of 15 rems (0.15 Sv), and

(ii) A shallow-dose equivalent of 50 rems (0.50 Sv) to the skin or to any extremity.

(b) Doses received in excess of the annual limits, including doses received during accidents, emergencies, and planned special exposures, must be subtracted from the limits for planned special exposures that the individual may receive during the current year (see §20.1206(e)(1)) and during the individual's lifetime (see §20.1206(e)(2)).

(c) The assigned deep-dose equivalent and shallow-dose equivalent must be for the part of the body receiving the highest exposure. The deep-dose equivalent, eye dose equivalent and shallow-dose equivalent may be assessed from surveys or other radiation measurements for the purpose of demonstrating compliance with the occupational dose limits, if the individual monitoring device was not in the region of highest potential exposure, or the results of individual monitoring are unavailable.

(d) Derived air concentration (DAC) and annual limit on intake (ALI) values are presented in table 1 of Appendix B to Part 20 and may be used to determine the individual's dose (see §20.2106) and to demonstrate compliance with the occupational dose limits.

(e) In addition to the annual dose limits, the licensee shall limit the soluble uranium intake by an individual to 10 milligrams in a week in consideration of chemical toxicity (see footnote 3 of appendix B to Part 20).

(f) The licensee shall reduce the dose that an individual may be allowed to receive in the current year by the amount of occupational dose received while employed by any other person (see §20.2104(e)).

20.1202 Compliance with requirements for summation of external and internal doses.

(a) If the licensee is required to monitor under both §20.1502(a) and (b), the licensee shall demonstrate compliance with the dose limits by summing external and internal doses. If the licensee is required to monitor only under §20.1502(a) or only under §20.1502(b), then summation is not required to demonstrate compliance with the dose limits. The licensee may demonstrate compliance with the requirements for summation of external and internal doses by meeting one of the conditions specified in paragraph (b) of this section and the conditions in paragraphs (c) and (d) of this section.

(Note: The dose equivalents for the lens of the eye, the skin, and the extremities are not included in the summation, but are subject to separate limits.)

(b) Intake by inhalation. If the only intake of radionuclides is by inhalation, the total effective dose equivalent limit is not exceeded if the sum of the deep-dose equivalent divided by the total effective dose equivalent limit, and one of the following, does not exceed unity:

(1) The sum of the fractions of the inhalation ALI for each radionuclide, or

(2) The total number of derived air concentration-hours (DAC-hours) for all radionuclides divided by 2,000, or

(3) The sum of the calculated committed effective dose equivalents to all significantly irradiated[1] organs or tissues (T) calculated from bioassay data using appropriate biological models and expressed as a fraction of the annual limit.

(c) Intake by oral ingestion. If the occupationally exposed individual also receives an intake of radionuclides by oral ingestion greater than 10 percent of the applicable oral ALI, the licensee shall account for this intake and include it in demonstrating compliance with the limits.

[1]An organ or tissue is deemed to be significantly irradiated if, for that organ or tissue, the product of the weighting factor, w_T, and the committed dose equivalent, $H_{T,50}$, per unit intake is greater than 10 percent of the maximum weighted value of $H_{T,50}$, that is W_T $H_{T,50}$ per unit intake for any organ or tissue.

(d) Intake through wounds or absorption through skin. The licensee shall evaluate and, to the extent practical, account for intakes through wounds or skin absorption.

Note: The intake through intact skin has been included in the calculation of DAC for hydrogen-3 and does not need to be further evaluated.

20.1203 Determination of external dose from airborne radioactive material.

Licensees shall, when determining the dose from airborne radioactive material, include the contribution to the deep-dose equivalent, eye dose equivalent, and shallow-dose equivalent from external exposure to the radioactive cloud (see Appendix B to Part 20 footnotes 1 and 2).

Note: Airborne radioactivity measurements and DAC values should be used as the primary means to assess the deep-dose equivalent when the airborne radioactive material includes radionuclides other than noble gases or if the cloud of airborne radioactive material is not relatively uniform. The determination of the deep-dose equivalent to an individual should be based upon measurements using instruments or individual monitoring devices.

20.1204 Determination of internal exposure.

(a) For purposes of assessing dose used to determine compliance with occupational dose equivalent limits, the licensee shall, when required under §20.1502, take suitable and timely measurements of-

(1) Concentrations of radioactive materials in air in work areas; or

(2) Quantities of radionuclides in the body; or

(3) Quantities of radionuclides excreted from the body; or

(4) Combinations of these measurements.

(b) Unless respiratory protective equipment is used, as provided in §20.1703, or the assessment of intake is based on bioassays, the licensee shall assume that an individual inhales radioactive material at the airborne concentration in which the individual is present.

(c) When specific information on the physical and biochemical properties of the radionuclides taken into the body or the behavior or the material in an individual is known, the licensee may-

(1) Use that information to calculate the committed effective dose equivalent, and, if used, the licensee shall document that information in the individual's record; and

(2) Upon prior approval of the Commission, adjust the DAC or ALI values to reflect the actual physical and chemical

characteristics of airborne radioactive material such as aerosol size distribution or density; and

(3) Separately assess the contribution of fractional intakes of Class D, W, or Y compounds of a given radionuclide (see appendix B to Part 201) to the committed effective dose equivalent.

(d) If the licensee chooses to assess intakes of Class Y material using the measurements given in §20.1204(a)(2) or (3), the licensee may delay the recording and reporting of the assessments for periods up to 7 months, unless otherwise required by §20.2202 or §20.2203, in order to permit the licensee to make additional measurements basic to the assessments.

(e) If the identity and concentration of each radionuclide in a mixture are known, the fraction of the DAC applicable to the mixture for use in calculating DAC-hours must be either-

(1) The sum of the ratios of the concentration to the appropriate DAC value such as D, W, Y from Appendix B to Part 20 for each radio-nuclide in the mixture; or

(2) The ratio of the total concentration for all radionuclides in the mixture to the most restrictive DAC value for any radionuclide in the mixture.

(f) If the identity of each radionuclide in a mixture is known, but the concentration of one or more of the radionuclides in the mixture is not known, the DAC for the mixture must be the most restrictive DAC of any radionuclide in the mixture.

(g) When a mixture of radionuclides in air exists, licensees may disregard certain radionuclides in the mixture if-

(1) The licensee uses the total activity of the mixture in demonstrating compliance with the dose limits in §20.1201 and in complying with the monitoring requirements in §20.1502(b), and

(2) The concentration of any radionuclide disregarded is less than 10 percent of its DAC, and

(3) The sum of these percentages for all of the radionuclides disregarded in the mixture does not exceed 30 percent.

(h)(1) In order to calculate the committed effective dose equivalent, the licensee may assume that the inhalation of one ALI, or an exposure of 2,000 DAC-hours, results in a committed effective dose equivalent of 5 rems (0.05 Sv) for radionuclides that have their ALIs or DACs based on the committed effective dose equivalent.

(2) When the ALI (and the associated DAC) is determined by the nonstochastic organ dose limit of 50 rems (0.5 Sv), the intake of radionuclides that would result in a committed effective dose equivalent of 5 rems (0.05 Sv) (the stochastic ALI) is listed in parentheses in table 1 of Appendix B to Part 20. In this case, the licensee may, as a simplifying assumption, use the stochastic ALIs

to determine committed effective dose equivalent. However, if the licensee uses the stochastic ALIs, the licensee must also demonstrate that the limit in §20.1201(a)(1)(ii) is met.

20.1206 Planned special exposures.

A licensee may authorize an adult worker to receive doses in addition to and accounted for separately from the doses received under the limits specified in §20.1201 provided that each of the following conditions is satisfied-

(a) The licensee authorizes a planned special exposure only in an exceptional situation when alternatives that might avoid the higher exposure are unavailable or impractical.

(b) The licensee (and employer if the employer is not the licensee) specifically authorizes the planned special exposure, in writing, before the exposure occurs.

(c) Before a planned special exposure, the licensee ensures that the individuals involved are-

(1) Informed of the purpose of the planned operation;

(2) Informed of the estimated doses and associated potential risks and specific radiation levels or other conditions that might be involved in performing the task; and

(3) Instructed in the measures to be taken to keep the dose ALARA considering other risks that may be present.

(d) Prior to permitting an individual to participate in a planned special exposure, the licensee ascertains prior doses as required by §20.2104(b) during the lifetime of the individual for each individual involved.

(e) Subject to §20.1201(b), the licensee does not authorize a planned special exposure that would cause an individual to receive a dose from all planned special exposures and all doses in excess of the limits to exceed-

(1) The numerical values of any of the dose limits in §20.1201(a) in any year; and

(2) Five times the annual dose limits in §20.1201(a) during the individual's lifetime.

(f) The licensee maintains records of the conduct of a planned special exposure in accordance with §20.2105 and submits a written report in accordance with §20.2204.

(g) The licensee records the best estimate of the dose resulting from the planned special exposure in the individual's record and informs the individual, in writing, of the dose within 30 days from the date of the planned special exposure. The dose from planned special exposures is not to be considered in controlling future occupational dose of the individual under §20.1201(a) but is to be included in evaluations required by §20.1206 (d) and (e).

20.1207 Occupational dose limits for minors.

The annual occupational dose limits for minors are 10 percent of the annual dose limits specified for adult workers in §20.1201.

20.1208 Dose to an embryo/fetus.

(a) The licensee shall ensure that the dose to an embryo/fetus during the entire pregnancy, due to occupational exposure of a declared pregnant woman, does not exceed 0.5 rem (5 mSv). (For recordkeeping requirements, see §20.2106.)

(b) The licensee shall make efforts to avoid substantial variation above a uniform monthly exposure rate to a declared pregnant woman so as to satisfy the limit in paragraph (a) of this section.

(c) The dose to an embryo/fetus shall be taken as the sum of-

(1) The deep-dose equivalent to the declared pregnant woman; and

(2) The dose to the embryo/fetus from radionuclides in the embryo/fetus and radionuclides in the declared pregnant woman.

(d) If the dose to the embryo/fetus is found to have exceeded 0.5 rem (5 mSv), or is within 0.05 rem (0.5 mSv) of this dose, by the time the woman declares the pregnancy to the licensee, the licensee shall be deemed to be in compliance with paragraph (a) of this section if the additional dose to the embryo/fetus does not exceed 0.05 rem (0.5 mSv) during the remainder of the pregnancy.

Subpart D - Radiation Dose Limits for Individual Member of the Public

20.1302 Compliance with dose limits for individual members of the public.

(a) The licensee shall make or cause to be made, as appropriate, surveys of radiation levels in unrestricted and controlled areas and radioactive materials in effluents released to unrestricted and controlled areas to demonstrate compliance with the dose limits for individual members of the public in §20.1301.

(b) A licensee shall show compliance with the annual dose limit in §20.1301 by-

(1) Demonstrating by measurement or calculation that the total effective dose equivalent to the individual likely to receive the highest dose from the licensed operation does not exceed the annual dose limit; or

(2) Demonstrating that-

(i) The annual average concentrations of radioactive material released in gaseous and liquid effluents at the boundary of the unrestricted area do not exceed the values specified in table 2 of Appendix B to Part 20; and

(ii) If an individual were continuously present in an unrestricted area, the dose from external sources would not exceed 0.002 rem (0.02 mSv) in an hour and 0.05 rem (0.5 mSv) in a year.

Subpart F - Surveys and Monitoring

20.1501 General.

(a) Each licensee shall make or cause to be made, surveys that-

(1) May be necessary for the licensee to comply with the regulations in this part; and

(2) Are reasonable under the circumstances to evaluate-

(i) The extent of radiation levels; and

(ii) Concentrations or quantities of radioactive material; and

(iii) The potential radiological hazards that could be present.

(b) The licensee shall ensure that instruments and equipment used for quantitative radiation measurements such as dose rate and effluent monitoring are calibrated periodically for the radiation measured.

(c) All personnel dosimeters (except for direct and indirect reading pocket ionization chambers and those dosimeters used to measure the dose to the extremities) that require processing to determine the radiation dose and that are used by licensees to comply with §20.1201, with other applicable provisions of this chapter, or with conditions specified in a license must be processed and evaluated by a dosimetry processor-

(1) Holding current personnel dosimetry accreditation from the National Voluntary Laboratory Accreditation Program (NVLAP) of the National Institute of Standards and Technology; and

(2) Approved in this accreditation process for the type of radiation or radiations included in the NVLAP program that most closely approximates the type of radiation or radiations for which the individual wearing the dosimeter is monitored.

20.1502 Conditions requiring individual monitoring of external and internal occupational dose.

Each licensee shall monitor exposures to radiation and radioactive material at levels sufficient to demonstrate compliance with the occupational dose limits of this part. As a minimum-

(a) Each licensee shall monitor occupational exposure to radiation and shall supply and require the use of individual monitoring devices by-

(1) Adults likely to receive, in 1 year from sources external to the body, a dose in excess of 10 percent of the limits in §20.1201(a),

(2) Minors and declared pregnant women likely to receive, in 1 year from sources external to the body, a dose in excess of 10

percent of any of the applicable limits in §20.1207 or §20.1208, and

(3) Individuals entering a high or very high radiation area.

(b) Each licensee shall monitor (see §20.1204) the occupational intake of radioactive material by and assess the committed effective dose equivalent to-

(1) Adults likely to receive, in 1 year, an intake in excess of 10 percent of the applicable ALI(s) in table 1, Columns 1 and 2, of Appendix B to Part 20; and

(2) Minors and declared pregnant women likely to receive, in 1 year, a committed effective dose equivalent in excess of 0.05 rem (0.5 mSv).

Subpart G - Control of Exposure From External Sources in Restricted Areas

20.1601 Control of access to high radiation areas.

(a) The licensee shall ensure that each entrance or access point to a high radiation area has one or more of the following features-

(1) A control device that, upon entry into the area, causes the level of radiation to be reduced below that level at which an individual might receive a deep-dose equivalent of 0.1 rem (1 mSv) in 1 hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates;

(2) A control device that energizes a conspicuous visible or audible alarm signal so that the individual entering the high radiation area and the supervisor of the activity are made aware of the entry; or

(3) Entryways that are locked, except during periods when access to the areas is required, with positive control over each individual entry.

(b) In place of the controls required by paragraph (a) of this section for a high radiation area, the licensee may substitute continuous direct or electronic surveillance that is capable of preventing unauthorized entry.

(c) A licensee may apply to the Commission for approval of alternative methods for controlling access to high radiation areas.

(d) The licensee shall establish the controls required by paragraphs (a) and (c) of this section in a way that does not prevent individuals from leaving a high radiation area.

(e) Control is not required for each entrance or access point to a room or other area that is a high radiation area solely because of the presence of radioactive materials prepared for transport and packaged and labeled in accordance with the regulations of the Department of Transportation provided that-

(1) The packages do not remain in the area longer than 3 days; and

(2) The dose rate at 1 meter from the external surface of any package does not exceed 0.01 rem (0.1 mSv) per hour.

20.1602 Control of access to very high radiation areas.

In addition to the requirements in §20.1601, the licensee shall institute additional measures to ensure that an individual is not able to gain unauthorized or inadvertent access to areas in which radiation levels could be encountered at 500 rads (5 grays) or more in 1 hour at 1 meter from a radiation source or any surface through which the radiation penetrates.

Subpart H - Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas

20.1701 Use of process or other engineering controls.

The licensee shall use, to the extent practical, process or other engineering controls such as containment or ventilation to control the concentrations of radioactive material in air.

20.1702 Use of other controls.

When it is not practical to apply process or other engineering controls to control the concentrations of radioactive material in air to values below those that define an airborne radioactivity area, the licensee shall, consistent with maintaining the total effective dose equivalent ALARA, increase monitoring and limit intakes by one or more of the following means:

(a) Control of access;

(b) Limitation of exposure times;

(c) Use of respiratory protection equipment; or

(d) Other controls.

20.1703 Use of individual respiratory protection equipment.

(a) If the licensee uses respiratory protection equipment to limit intakes pursuant to §20.1702-

(1) The licensee shall use only respiratory protection equipment that is tested and certified or had certification extended by the National Institute for Occupational Safety and Health/Mine Safety and Health Administration (NIOSH/MSHA).

(2) If the licensee wishes to use equipment that has not been tested or certified by NIOSH/MSHA, has not had certification extended by NIOSH/MSHA, or for which there is no schedule for testing or certification, the licensee shall submit an application for authorized use of that equipment, including a demonstration by testing, or a demonstration on the basis of reliable test information, that the material

and performance characteristics of the equipment are capable of providing the proposed degree of protection under anticipated conditions of use.

(3) The licensee shall implement and maintain a respiratory protection program that includes-

(i) Air sampling sufficient to identify the potential hazard, permit proper equipment selection, and estimate exposures;

(ii) Surveys and bioassays, as appropriate, to evaluate actual intakes;

(iii) Testing of respirators for operability immediately prior to each use;

(iv) Written procedures regarding selection, fitting, issuance, maintenance, and testing of respirators, including testing for operability immediately prior to each use; supervision and training of personnel; monitoring, including air sampling and bioassays; and recordkeeping; and

(v) Determination by a physician prior to the initial fitting of respirators, and either every 12 months thereafter or periodically at a frequency determined by a physician, that the individual user is medically fit to use the respiratory protection equipment.

(4) The licensee shall issue a written policy statement on respirator usage covering-

(i) The use of process or other engineering controls, instead of respirators;

(ii) The routine, nonroutine, and emergency use of respirators; and

(iii) The periods of respirator use and relief from respirator use.

(5) The licensee shall advise each respirator user that the user may leave the area at any time for relief from respirator use in the event of equipment malfunction, physical or psychological distress, procedural or communication failure, significant deterioration of operating conditions, or any other conditions that might require such relief.

(6) The licensee shall use equipment within limitations for type and mode of use and shall provide proper visual, communication, and other special capabilities (such as adequate skin protection) when needed.

(b) In estimating exposure of individuals to airborne radioactive materials, the licensee may make allowance for respiratory protection equipment used to limit intakes pursuant to §20.1702, provided that the following conditions, in addition to those in §20.1703(a), are satisfied:

(1) The licensee selects respiratory protection equipment that provides a protection factor (see Appendix A, Part 20) greater than the multiple by which peak concentrations of airborne radioactive materials in the working area are expected to exceed the values specified in Appendix B to Part 20, Table 1, column 3. If the selection of a respiratory protection device with a protection factor greater than the multiple defined in the preceding sentence is

inconsistent with the goal specified in §20.1702 of keeping the total effective dose equivalent ALARA, the licensee may select respiratory protection equipment with a lower protection factor only if such a selection would result in keeping the total effective dose equivalent ALARA. The concentration of radioactive material in the air that is inhaled when respirators are worn may be initially estimated by dividing the average concentration in air, during each period of uninterrupted use, by the protection factor. If the exposure is later found to be greater than estimated, the corrected value must be used; if the exposure is later found to be less than estimated, the corrected value may be used.

(2) The licensee shall obtain authorization from the Commission before assigning respiratory protection factors in excess of those specified in Appendix A to Part 20. The Commission may authorize a licensee to use higher protection factors on receipt of an application that-

(i) Describes the situation for which a need exists for higher protection factors, and

(ii) Demonstrates that the respiratory protection equipment provides these higher protection factors under the proposed conditions of use.

(c) The licensee shall use as emergency devices only respiratory protection equipment that has been specifically certified or had certification extended for emergency use by NIOSH/MSHA.

(d) The licensee shall notify, in writing, the Regional Administrator of the appropriate NRC Regional Office listed in Appendix D to Part 20 at least 30 days before the date that respiratory protection equipment is first used under the provisions of either §20.1703 (a) or (b).

Subpart K - Waste Disposal

20.2001 General requirements.

(a) A licensee shall dispose of licensed material only--

(1) By transfer to an authorized recipient as provided in §20.2006 or in the regulations in parts 30, 40, 60, 61, 70, or 72 of this chapter; or

(2) By decay in storage; or

(3) By release in effluents within the limits in 20.1301; or

(4) As authorized under §§20.2002, 20.2003, 20.2004, or 20.2005.

(b) A person must be specifically licensed to receive waste containing licensed material from other persons for:

(1) Treatment prior to disposal; or

(2) Treatment or disposal by incineration; or

(3) Decay in storage; or

(4) Disposal at a land disposal facility licensed under part 61 of this chapter; or

(5) Disposal at a geologic repository under part 60 of this chapter.

20.2002 Method for obtaining approval of proposed disposal procedures.

A licensee or applicant for a license may apply to the Commission for approval of proposed procedures not otherwise authorized in the regulations in this chapter, to dispose of licensed material generated in the licensee's activities. Each application shall include: (a) A description of the waste containing licensed material to be disposed of, including the physical and chemical properties important to risk evaluation, and the proposed manner and conditions of waste disposal; and

(b) An analysis and evaluation of pertinent information on the nature of the environment; and

(c) The nature and location of other potentially affected licensed and unlicensed facilities; and

(d) Analyses and procedures to ensure that doses are maintained ALARA and within the dose limits in this part.

20.2003 Disposal by release into sanitary sewerage.

(a) A licensee may discharge licensed material into sanitary sewerage if each of the following conditions is satisfied:

(1) The material is readily soluble (or is readily dispersible biological material) in water; and

(2) The quantity of licensed or other radioactive material that the licensee releases into the sewer in 1 month divided by the average monthly volume of water released into the sewer by the licensee does not exceed the concentration listed in table 3 of Appendix B to Part 20; and

(3) If more than one radionuclide is released, the following conditions must also be satisfied:

(i) The licensee shall determine the fraction of the limit in table 3 of appendix B to §§20.1001-20.2401 represented by discharges into sanitary sewerage by dividing the actual monthly average concentration of each radionuclide released by the licensee into the sewer by the concentration of that radionuclide listed in table 3 of Appendix B to Part 20; and

(ii) The sum of the fractions for each radionuclide required by paragraph (a)(3)(i) of this section does not exceed unity; and

(4) The total quantity of licensed and other radioactive material that the licensee releases into the sanitary sewerage system in a year does not exceed 5 curies (185 GBq) of hydrogen-

3, 1 curie (37 GBq) of carbon-14, and 1 curie (37 GBq) of all other radioactive materials combined.

(b) Excreta from individuals undergoing medical diagnosis or therapy with radioactive material are not subject to the limitations contained in paragraph (a) of this section.

20.2004 Treatment or disposal by incineration.

(a) A licensee may treat or dispose of licensed material by incineration only;

(1) As authorized by paragraph (b) of this section; or

(2) If the material is in a form and concentration specified in 20.2005; or

(3) As specifically approved by the Commission pursuant to 20.2002.

(b)(1) Waste oils (petroleum derived or synthetic oils used principally as lubricants, coolants, hydraulic or insulating fluids, or metalworking oils) that have been radioactively contaminated in the course of the operation or maintenance of a nuclear power reactor licensed under part 50 of this chapter may be incinerated on the site where generated provided that the total radioactive effluents from the facility, including the effluents from such incineration, conform to the requirements of Appendix I to Part 50 of this chapter and the effluent release limits contained in applicable license conditions other than effluent limits specifically related to the incineration of waste oil. The licensee shall report any changes or additions to the information supplied under 50.34 and §50.34e of this chapter associated with this incineration pursuant to 50.71 of this chapter, as appropriate. The licensee shall also follow the procedures of §50.59 of this chapter with respect to such changes to the facility or procedures.

(2) Solid residues produced in the process of incinerating waste oils must be disposed of as provided by §20.2001.

(3) The provisions of this section authorize onsite waste oil incineration under the terms of this section and supersede any provision in an individual plant license or technical specification that may be inconsistent.

20.2005 Disposal of specific wastes.

(a) A licensee may dispose of the following licensed material as if it were not radioactive:

(1) 0.05 microcurie (1.85 kBq), or less, of hydrogen-3 or carbon-14 per gram of medium used for liquid scintillation counting; and

(2) .05 microcurie (1.85 kBq), or less, of hydrogen-3 or carbon-14 per gram of animal tissue, averaged over the weight of the entire animal.

(b) A licensee may not dispose of tissue under paragraph (a)(2) of this section in a manner that would permit its use either as food for humans or as animal feed.

(c) The licensee shall maintain records in accordance with 20.2108.

20.2006 Transfer for disposal and manifests.

(a)(1) The requirements of this section and appendices F and G to 10 CFR part 20 are designed to

(i) Control transfers of low-level radioactive waste by any waste generator, waste collector, or waste processor licensee, as defined in this part, who ships low-level waste either directly, or indirectly through a waste collector or waste processor, to a licensed low-level waste land disposal facility (as defined in part 61 of this chapter);

(ii) Establish a manifest tracking system; and

(iii) Supplement existing requirements concerning transfers and recordkeeping for those wastes.

(2) Beginning March 1, 1998, all affected licensees must use Appendix G. Prior to March 1, 1998, a LLW disposal facility operator or its regulatory authority may require the shipper to use appendix F or appendix G. Licensees using appendix F shall comply with paragraph (b)(1) of this section. Licensees using appendix G shall comply with paragraph (b)(2) of this section.

(b)(1) Each shipment of radioactive waste intended for disposal at a licensed land disposal facility must be accompanied by a shipment manifest in accordance with section I of appendix F to 10 CFR part 20.

(2) Any licensee shipping radioactive waste intended for ultimate disposal at a licensed land disposal facility must document the information required on NRC's Uniform Low-Level Radioactive Waste Manifest and transfer this recorded manifest information to the intended consignee in accordance with appendix G to 10 CFR part 20.

(c) Each shipment manifest must include a certification by the waste generator as specified in section II of appendix F or appendix G to 10 CFR part 20, as appropriate. See paragraph (a)(2) of this section to determine the appropriate appendix.

(d) Each person involved in the transfer for disposal and disposal of waste, including the waste generator, waste collector, waste processor, and disposal facility operator, shall comply with the requirements specified in section III of appendix F or appendix G to 10 CFR part 20, as appropriate. See paragraph (a)(2) of this section to determine the appropriate appendix.

20.2007 Compliance with environmental and health protection regulations.

Nothing in this subpart relieves the licensee from complying with other applicable Federal, State, and local regulations governing any other toxic or hazardous properties of materials that may be disposed of under this subpart.

Subpart L - Records

20.2102 Records of radiation protection programs.

(a) Each licensee shall maintain records of the radiation protection program,
including:...

(2) Audits and other reviews of program content and implementation.

20.2103 Records of surveys.

(a) Each licensee shall maintain records showing the results of surveys and calibrations required by §20.1501 and §20.1906(b). The licensee shall retain these records for 3 years after the record is made.

(b) The licensee shall retain each of the following records until the Commission terminates each pertinent license requiring the record:

(1) Records of the results of surveys to determine the dose from external sources and used, in the absence of or in combination with individual monitoring data, in the assessment of individual dose equivalents. This includes those records of results of surveys to determine the dose from external sources and used, in the absence of or in combination with individual monitoring data, in the assessment of individual dose equivalents required under the standards for protection against radiation in effect prior to January 1, 1994; and

(2) Records of the results of measurements and calculations used to determine individual intakes of radioactive material and used in the assessment of internal dose. This includes those records of the results of measurements and calculations used to determine individual intakes of radioactive material and used in the assessment of internal dose required under the standards for protection against radiation in effect prior to January 1, 1994; and

(3) Records showing the results of air sampling, surveys, and bioassays required pursuant to §20.1703(a)(3) (I) and (ii). This includes those records showing the results of air sampling, surveys, and bioassays required under the standards for protection against radiation in effect prior to January 1, 1994; and

(4) Records of the results of measurements and calculations used to evaluate the release of radioactive effluents to the

environment. This includes those records of the results of measurements and calculations used to evaluate the release of radioactive effluents to the environment required under the standards for protection against radiation in effect prior to January 1, 1994.

20.2104 Determination of prior occupational dose.

(a) For each individual who is likely to receive in a year, an occupational dose requiring monitoring pursuant to §20.1502 the licensee shall-

(1) Determine the occupational radiation dose received during the current year; and

(2) Attempt to obtain the records of cumulative occupational radiation dose.

(b) Prior to permitting an individual to participate in a planned special exposure, the licensee shall determine-

(1) The internal and external doses from all previous planned special exposures; and

(2) All doses in excess of the limits (including doses received during accidents and emergencies) received during the lifetime of the individual.

(c) In complying with the requirements of paragraph (a) of this section, a licensee may-

(1) Accept, as a record of the occupational dose that the individual received during the current year, a written signed statement from the individual, or from the individual's most recent employer for work involving radiation exposure, that discloses the nature and the amount of any occupational dose that the individual may have received during the current year;

(2) Accept, as the record of cumulative radiation dose, an up-to-date NRC Form 4, or equivalent, signed by the individual and countersigned by an appropriate official of the most recent employer for work involving radiation exposure, or the individual's current employer (if the individual is not employed by the licensee); and

(3) Obtain reports of the individual's dose equivalent(s) from the most recent employer for work involving radiation exposure, or the individual's current employer (if the individual is not employed by the licensee) by telephone, telegram, electronic media, or letter. The licensee shall request a written verification of the dose data if the authenticity of the transmitted report cannot be established.

(d) The licensee shall record the exposure history of each individual, as required by paragraph (a) of this section, on NRC Form 4, or other clear and legible record, including all of the

information required by NRC Form 4.[2] The form or record must show each period in which the individual received occupational exposure to radiation or radioactive material and must be signed by the individual who received the exposure. For each period for which the licensee obtains reports, the licensee shall use the dose shown in the report in preparing the NRC Form 4. For any period in which the licensee does not obtain a report, the licensee shall place a notation on the NRC Form 4 indicating the periods of time for which data are not available.

(e) If the licensee is unable to obtain a complete record of an individual's current and previously accumulated occupational dose, the licensee shall assume-

(1) In establishing administrative controls under §20.1201(f) for the current year, that the allowable dose limit for the individual is reduced by 1.25 rems (12.5 mSv) for each quarter for which records were unavailable and the individual was engaged in activities that could have resulted in occupational radiation exposure; and

(2) That the individual is not available for planned special exposures.

(f) The licensee shall retain the records on NRC Form 4 or equivalent until the Commission terminates each pertinent license requiring this record. The licensee shall retain records used in preparing NRC Form 4 for 3 years after the record is made. This includes records required under the standards for protection against radiation in effect prior to January 1, 1994.

20.2105 Records of planned special exposures.

(a) For each use of the provisions of §20.1206 for planned special exposures, the licensee shall maintain records that describe-

(1) The exceptional circumstances requiring the use of a planned special exposure; and

(2) The name of the management official who authorized the planned special exposure and a copy of the signed authorization; and

(3) What actions were necessary; and

(4) Why the actions were necessary; and

(5) How doses were maintained ALARA; and

[2] Licensees are not required to partition historical dose between external dose equivalent(s) and internal committed dose equivalent(s). Further, occupational exposure histories obtained and recorded on NRC Form 4 before January 1, 1994, might not have included effective dose equivalent, but may be used in the absence of specific information on the intake of radionuclides by the individual.

(6) What individual and collective doses were expected to result, and the doses actually received in the planned special exposure.

(b) The licensee shall retain the records until the Commission terminates each pertinent license requiring these records.

20.2106 Records of individual monitoring results.

(a) Recordkeeping requirement. Each licensee shall maintain records of doses received by all individuals for whom monitoring was required pursuant to §20.1502, and records of doses received during planned special exposures, accidents, and emergency conditions. These records[3] must include, when applicable-

(1) The deep-dose equivalent to the whole body, eye dose equivalent, shallow-dose equivalent to the skin, and shallow-dose equivalent to the extremities; and

(2) The estimated intake or body burden of radionuclides (see §20.1202); and

(3) The committed effective dose equivalent assigned to the intake or body burden of radionuclides; and

(4) The specific information used to calculate the committed effective dose equivalent pursuant to §20.1204(c); and

(5) The total effective dose equivalent when required by §20.1202; and

(6) The total of the deep-dose equivalent and the committed dose to the organ receiving the highest total dose.

(b) Recordkeeping frequency. The licensee shall make entries of the records specified in paragraph (a) of this section at least annually.

(c) Recordkeeping format. The licensee shall maintain the records specified in paragraph (a) of this section on NRC Form 5, in accordance with the instructions for NRC Form 5, or in clear and legible records containing all the information required by NRC Form 5.

(d) Privacy protection. The records required under this section should be protected from public disclosure because of their personal privacy nature. These records are protected by most State privacy laws and, when transferred to the NRC, are protected by the Privacy Act of 1974, Public Law 93-579, 5 U.S.C. 552a, and the Commission's regulations in 10 CFR Part 9.

(e) The licensee shall maintain the records of dose to an embryo/fetus with the records of dose to the declared pregnant

[3]Assessments of dose equivalent and records made using units in effect before the licensee's adoption of this part need not be changed.

woman. The declaration of pregnancy shall also be kept on file, but may be maintained separately from the dose records.

(f) The licensee shall retain the required form or record until the Commission terminates each pertinent license requiring this record. This includes records required under the standards for protection against radiation in effect prior to January 1, 1994.

Subpart M - Reports

20.2203 Reports of exposures, radiation levels, and concentrations.

(a) Reportable events. In addition to the notification required by §20.2202, each licensee shall submit a written report within 30 days after learning of any of the following occurrences:

(1) Any incident for which notification is required by §20.2202; or

(2) Doses in excess of any of the following:

(i) The occupational dose limits for adults in §20.1201; or

(ii) The occupational dose limits for a minor in §20.1207; or

(iii) The limits for an embryo/fetus of a declared pregnant woman in §20.1208; or

(iv) The limits for an individual member of the public in §20.1301; or

(v) Any applicable limit in the license; or

(vi) The ALARA constraints for air emissions established under §20.1101(d); or

(3) Levels of radiation or concentrations of radioactive material in-

(i) A restricted area in excess of any applicable limit in the license; or

(ii) An unrestricted area in excess of 10 times any applicable limit set forth in this part or in the license (whether or not involving exposure of any individual in excess of the limits in §20.1301); or

(4) For licensees subject to the provisions of EPA's generally applicable environmental radiation standards in 40 CFR part 190, levels of radiation or releases of radioactive material in excess of those standards, or of license conditions related to those standards.

(b) Contents of reports. (1) Each report required by paragraph (a) of this section must describe the extent of exposure of individuals to radiation and radioactive material, including, as appropriate:

(i) Estimates of each individual's dose; and

(ii) The levels of radiation and concentrations of radioactive material involved; and

(iii) The cause of the elevated exposures, dose rates, or concentrations; and

(iv) Corrective steps taken or planned to ensure against a recurrence, including the schedule for achieving conformance with applicable limits, ALARA constraints, generally applicable environmental standards, and associated license conditions.

(2) Each report filed pursuant to paragraph (a) of this section must include for each occupationally overexposed[4] individual: the name, Social Security account number, and date of birth. The report must be prepared so that this information is stated in a separate and detachable part of the report.

(c) For holders of an operating license for a nuclear power plant, the occurrences included in paragraph (a) of this section must be reported in accordance with the procedures described in §50.73(b), (c), (d), (e), and (g) of this chapter and must also include the information required by paragraph (b) of this section. Occurrences reported in accordance with §50.73 of this chapter need not be reported by a duplicate report under paragraph (a) of this section.

(d) All licensees, other than those holding an operating license for a nuclear power plant, who make reports under paragraph (a) of this section shall submit the report in writing to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, U.S. 20555, with a copy to the appropriate NRC Regional Office listed in appendix D to Part 20.

10 CFR Part 60, Disposal of High-Level Radioactive Wastes in Geologic Repositories
Subpart B - Licenses
License Applications
60.21 Content of application.

(c) The Safety Analysis Report shall include...

(3) A description and analysis of the design and performance requirements for structures, systems, and components of the geologic repository that are important to safety. The analysis must include a demonstration that--

(i) The requirements of §60.111(a) will be met, assuming occurrence of Category 1 design basis events...

(7) A description of the program for control and monitoring of radioactive effluents and occupational radiation exposures to maintain such effluents and exposures in accordance with the requirements of part 20 of this chapter....

[4] With respect to the limit for the embryo-fetus (§20.1208), the identifiers should be those of the declared pregnant woman.

(15) The following information concerning activities at the geologic repository operations area:...

(v) Plans for conduct of normal activities, including maintenance, surveillance, and periodic testing of structures, systems, and components of the geologic repository operation area.

(vi) Plans for permanent closure and plans for the decontamination or dismantlement of surface facilities.

Construction Authorization.

60.31 Construction authorization.

Upon review and consideration of an application and environmental impact statement submitted under this part, the Commission may authorize construction if it determines:

(a) Safety. That there is reasonable assurance that the types and amounts of radioactive materials described in the application can be received, possessed, and disposed of in a geologic repository operations area of the design proposed without unreasonable risk to the health and safety of the public. In arriving at this determination, the Commission shall consider whether:

(1) The DOE has described the proposed geologic repository including but not limited to:... (ii) the kinds and quantities of radioactive waste to be received, possessed, stored, and disposed of in the geologic repository operations area...

Subpart E - Technical Criteria

60.101 Purpose and nature of findings.

(b) Subpart B of this part also lists findings that must be made in support of an authorization to construct a geologic repository operations area. In particular, §60.31(a) requires a finding that there is reasonable assurance that the types and amounts of radioactive materials described in the application can be received, possessed, and disposed of in a geologic repository operations area of the design proposed without unreasonable risk to the health and safety of the public. As stated in that paragraph, in arriving at this determination, the Commission will consider whether the site and design comply with the criteria contained in this subpart...

Performance Objectives

60.111 Performance of the geologic repository operations area through permanent closure.

(a) *Protection against radiation exposures and releases of radioactive material.* The geologic repository operations area shall be designed so that until permanent closure has been completed, radiation exposures and radiation levels, and releases of radioactive materials to unrestricted areas, will be maintained within the limits specified in part 20 of this chapter and such

generally applicable environmental standards for radioactivity as may have been established by Environmental Protection Agency.

Design Criteria for the Geologic Repository Operations Area

60.131 General design criteria for the geologic repository operations area.

(a) *Radiological protection.* The geologic repository operations area shall be designed to maintain radiation doses, levels, and concentrations of radioactive material in air in restricted areas within the limits specified in part 20 of this chapter. Design shall include:

(1) Means to limit concentrations of radioactive material in air;

(2) Means to limit the time required to perform work in the vicinity of radioactive materials, including, as appropriate, designing equipment for ease of repair and replacement and providing adequate space for ease of operation;

(3) Suitable shielding;

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

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

(6) A radiation alarm system to warn of significant increases in radiation levels, concentrations of radioactive material in air, and of increased radioactivity released in effluents. The alarm system shall be designed with provisions for calibration and for testing its operability.

60.132 Additional design criteria for surface facilities in the geologic repository operations area.

(b) Surface facility ventilation. Surface facility ventilation systems supporting waste transfer, inspection, decontamination, processing, or packaging shall be designed to provide protection against radiation exposures and offsite releases as provided in §60.111(a).

60.133 Additional design criteria for the underground facility.

(g) Underground facility ventilation. The ventilation system shall be designed to:

(1) Control the transport of radioactive particulates and gases within and releases from the underground facility in accordance with the performance objectives of §60.111(a).

(3) Separate the ventilation of excavation and waste emplacement areas.

IV. ACCEPTANCE CRITERIA AND GUIDANCE

General Guidance

This section contains the guidance for preparation of the License Application to demonstrate compliance of the radiation protection program for the Monitored Geologic Repository (MGR).

Some guidance is provided in this chapter by reference to NRC regulatory guides, NUREG documents, and industry codes and standards. Unless otherwise noted in the references to these documents, applicability of these documents and how to address them has not been fully determined. A preliminary assessment of the applicability of the NRC regulatory guides has been completed and incorporated into this guidance. Other potentially applicable documents (for example, industry codes and standards, NUREG documents) that may be useful are identified. However, a formal assessment of the applicability of these additional guidance documents has not been incorporated into this version of the technical guidance document; applicable instructions for the License Application author will be provided in a future revision.

In general, the application for construction authorization must provide sufficient information so that NRC can make a determination that there is reasonable assurance that the types and amounts of radioactive materials described in the License Application can be received, possessed, and disposed of in the proposed Yucca Mountain geologic repository operations area without unreasonable risk to the health and safety of the public. To support that conclusion, this section of the License Application must provide the following information:

- Show that the site and design of the geologic repository operations area will allow radiation doses, radiation levels, and concentrations of radioactive material in air in restricted areas to be ALARA and maintained within the limits specified in 10 CFR 20, and that the design includes:
  - Means to limit concentrations of radioactive material in air
  - Means to limit the time required to perform work in the vicinity of radioactive materials, including, as appropriate, designing equipment for ease of repair and replacement and providing adequate space for ease of operation
  - Suitable shielding
  - Means to monitor and control the dispersal of radioactive contamination
  - Means to control access to high radiation areas or airborne radioactivity areas
  - An alarm system to warn of significant increases in radiation levels, concentrations of radioactive material in air, and of increased radioactivity released in effluents

- Describe the radiation protection program that will include policy considerations and management commitment to maintain occupational exposures ALARA and comply with the requirements of 10 CFR 20. Elements of the program include:
  - An operational health physics program
  - A low-level radioactive waste management program
  - A radiological environmental monitoring and surveillance program
- Describe the program for assessing doses to occupational workers, members of the public, and the environment. The dose assessment program should meet the requirements contained in 10 CFR 20.
- Describe the program for controlling the radiation doses to the public and the environment in accordance with the guidance in 10 CFR 20.
- Describe the program for radiation worker protection.
- Describe or cross-reference a description in another part of the License Application of the types and amounts of radioactive materials to be received, possessed, and disposed of in the proposed repository.

Chapter 10 of the License Application should be written in broadly descriptive terms without providing excessive detail. Identify and commit to applicable NRC guidance documents and standards that describe methods acceptable for satisfying regulatory requirements and contain information to support the licensing section. Reference other planned documents or manuals, such as an operational ALARA manual, as the documentation where programs or processes will be described in detail, with this technical guidance document providing an overview of those processes. Cases in which practices or features of the geologic repository operations area are substantially different from elsewhere in the nuclear industry (from a radiological protection perspective) require more detail. Cases in which practices are very similar to elsewhere in the industry require a lower level of detail. Cases in which industry standards are committed to, such as regulatory guides or nuclear guides, require little detail. Cases in which alternative approaches are to be used require a more extensive description. It is important that a sufficient level of detail be provided to give the NRC reviewers confidence that the project will implement a radiation protection program that will meet or exceed requirements.

Guidance in this chapter is based on, or extracted from, the following documents:

- NUREG-1323, *License Application Review Plan for a Geologic Repository for Spent Nuclear Fuel and High-Level Radioactive Waste* (NRC 1995)
- YMP/97-01Q, *Management Plan for the Development of a Project Integrated Safety Assessment (PISA)* (YMP 1997)
- NUREG-1567, *Standard Review Plan for Spent Fuel Dry Storage Facilities* (NRC 1996b)

- Regulatory Guide DG-3003, *Format and Content for the License Application for the High-Level Waste Repository* (NRC 1990)

NUREG-1323 (NRC 1995) is largely incomplete and will become inapplicable to Yucca Mountain upon the expected issuance of a new 10 CFR 63 as discussed in Section III above. Information from NUREG-1323 (NRC 1995) considered likely to be applicable under the new regulation is included in the technical guidance document, though the NUREG is not referenced as an authoritative source for such information. DG-3003 (NRC 1990) is a draft regulatory guide that will also become inapplicable to Yucca Mountain upon the expected issuance of a new 10 CFR 63 as discussed in Section III above. Information from DG-3003 (NRC 1990) considered likely to be applicable under the new regulation is included in the technical guidance document, though DG-3003 (NRC 1990) is not referenced as an authoritative source for such information. YMP/97-01Q (YMP 1997) contains the Project Integrated Safety Assessment Content Guide that has been developed by the Yucca Mountain Site Characterization Project. The Project Integrated Safety Assessment Content Guide is the starting point for guidance in this document. NUREG-1567 (NRC 1996b), though not directly applicable to the repository or the waste package, pertains to containers for dry storage of spent nuclear fuel and therefore contains much useful and transferable information for the License Application for the MGR. This document contains detailed information that is missing from other NRC guidance documents for the repository.

In addition to the documents identified in this section, the information in the *Mined Geologic Disposal System License Application Annotated Outline,* YMP/94-05 (DOE Office of Civilian Radioactive Waste Management, in draft), was used extensively to develop the topics and regulatory interfaces contained herein.

10.0 PURPOSE, SUMMARY, AND CONCLUSIONS

Provide the purpose of Chapter 10 of the License Application. Include a summary of the information contained in the chapter and a discussion of the organization of the chapter. Provide a summary of conclusions supporting the intent of this chapter and a description of important uncertainties, if applicable, that affect the conclusions.

10.1 RADIATION PROTECTION PROGRAMS

This section will describe management commitments to maintain occupational radiation doses ALARA and to comply with regulatory limits and constraints per NRC requirements. It will also describe elements of the operational health physics program that will combine to ensure occupational doses are maintained ALARA.

10.1.1 Policy Considerations

Describe the policies that will be implemented to maintain radiation doses ALARA and control contamination levels. Discuss who will establish ALARA goals and how goals will be set, reviewed, measured, and relayed to the workforce. Describe the operating philosophy that will be used to maintain the radiological safety of workers in the repository during the handling, storage, retrieval, emplacement, and isolation of radioactive waste packages for normal operations, anticipated operational occurrences, accident conditions, and performance

confirmation activities. Review Regulatory Guide 8.8, *Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable* (NRC 1978), and Regulatory Guide 8.10, *Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable* (NRC 1977), for applicability in implementing an ALARA program for the MGR. The project will assess the applicability of the guidance in these guides, and appropriate instructions for the License Application author will be provided in a future revision of the technical guidance document. Include the following in the policy discussion:

- Endorse the ALARA principle as a policy of management and engineering.
- Identify the goals and objectives of the ALARA program. Include the applicable elements in Regulatory Guide 8.8, Section C (NRC 1978), and Regulatory Guide 4.20, *Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees Other Than Power Reactors* (NRC 1996a).
- Describe the design and operational elements of management policy that will lead to maintaining doses ALARA and control contamination levels.
- Describe a program for the periodic review of the radiation protection and ALARA programs. Reviews should address program content and determine effectiveness in optimizing personnel doses.
- Describe a quality assurance corrective action process to be applied to the radiation protection program. Include provisions for escalation of issues and identification of corrective action plans in the description of the process.
- Describe a policy that is consistent with the guidance in Regulatory Guide 8.15, *Acceptable Programs for Respiratory Protection* (NRC 1976a, Section 3.31), to ensure compliance with 10 CFR 20.1703(a)(4)(i) through (iii). The policy will include the use of practicable engineering controls instead of respirators. The project will assess the applicability of the guidance in this regulatory guide, and appropriate instructions for the License Application author will be provided in a future revision of the technical guidance document.

10.1.2 Operational Health Physics

10 CFR 20.1101 requires that each licensee develop, document, and implement a radiation protection program commensurate with the scope and extent of licensed activities and sufficient to ensure compliance. The licensee is also required to use, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are ALARA. Describe a radiation protection program that will be developed, documented, and implemented to ensure compliance with the occupational dose limits and is ALARA as addressed in 10 CFR 20. This section will tie together ALARA principles with specific operational monitoring and protective activities that will be performed to ensure effective dose assessment and mitigation.

Describe each of the topics identified in enough detail to provide sufficient evidence that a fully compliant program will be in place by the time of issuance of a license to receive and possess high-level radioactive waste. Under each topic, discuss how the procedures in this program are intended to maintain radiation doses ALARA and control contamination levels. Include in each section a brief discussion of how the identified practices contribute to control and reduction of occupational dose. Describe in general terms the operational health physics program and associated practices that will implement ALARA principles. Incorporate guidance from applicable regulatory documents into appropriate section discussions.

Describe the administrative organization of the health physics program, including the authority and responsibility of the health physics organization and the management chain for radiation safety. Describe the authority and responsibility of each position identified, including identification of radiation protection staff to implement ALARA principles, and the activities to be conducted by the individuals having responsibility for radiation protection. The list of regulatory guides below should be considered for applicability to characterize an effective organization, including responsibilities and reporting relationships for a health physics program. Review these documents; some of their guidance is potentially applicable. The project will assess the applicability of the guidance in these documents and appropriate instructions for the License Application author will be provided in a future revision of the technical guidance document.

- Regulatory Guide 8.2, *Guide for Administrative Practices in Radiation Monitoring* (NRC 1973)
- Regulatory Guide 8.8, Positions 1 and 2, *Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable (ALARA)* (NRC 1978)
- Regulatory Guide 8.10, *Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable* (NRC 1977)

Cross-reference License Application Section 11.2 for description of the MGR organizational structure, management staffing, and administrative goals, and Section 11.8 for detailed discussion of staff training, qualification, and certification.

Describe how management support and commitment to radiation safety will be ensured. Relate this activity to the performance of audits in the area of radiation protection, and identify associated criteria. Include the guidance provided by Regulatory Guide 8.10, Regulatory Position 1 (NRC 1977).

Describe the organizational relationships between health physics, operations, planning, engineering, design, performance confirmation, and other groups for the implementation of ALARA processes. Include the guidance provided by Regulatory Guide 8.8, Regulatory Position 3 (NRC 1978), for operational considerations in terms of preparation and planning, operations, and post-operations activities. For each of these activities, briefly describe the philosophy of integrating the ALARA program with all applicable facility activities, including procedure usage, job planning, record keeping, special equipment, operating philosophy, access

control, work permits, modifications, and other support functions. For each of these, provide specific examples of actions that will be taken to demonstrate this integration.

In general terms, describe the facilities and equipment expected to be available for radiation protection purposes. Describe portable and laboratory facilities and equipment for radioactive material analyses, laboratory facilities and equipment for handling and for performing analyses of environmental samples, environmental monitoring and sampling equipment and facilities, area radiation monitoring, airborne sampling equipment, protective clothing, respiratory protective equipment, decontamination facilities for equipment and personnel, and other contamination control equipment availability. Describe the distinction between facilities for normal operation, anticipated operational occurrences, and accident conditions, as applicable.

Address the relationship between the design and location of the facilities and ALARA principles, as well as any other special environmental considerations that may be applicable, such as low background area for laboratory facilities and segregation of sample types to prevent cross contamination. Include the following contamination control equipment/areas that may be available: locker and shower rooms; offices; access control stations; laboratory facilities for radioactivity analyses; protective clothing and change facilities; respiratory protective equipment facilities; decontamination facilities for equipment and personnel; instrument storage, calibration, and maintenance facilities; and equipment to facilitate communications.

In discussion of facilities and equipment, indicate how the program will follow the guidance provided in Regulatory Guide 8.8, Regulatory Position 4 (NRC 1978).

Describe the portable and laboratory instrumentation that will be used for radiation and contamination surveys and airborne radioactivity monitoring and sampling; monitoring area radiation; and monitoring of personnel during normal operation, anticipated operational occurrences, and accident situations. Include the types and ranges of detectors and monitors in the descriptions. Describe the program to ensure that sufficient quantities of each type of equipment will be available for normal and abnormal situations.

10 CFR 20.1501(b) requires the licensee to ensure that instruments and equipment used for quantitative radiation measurements are calibrated periodically for the radiation measured. Discuss the methods that will be in place to ensure instrument and measurement accuracy and precision for radiation survey instrumentation, laboratory instrumentation, and for instrumentation that will be used for environmental monitoring. Identify calibration frequencies for the different types of instruments or, alternatively, briefly discuss how the frequencies will be determined.

10 CFR 20.1101(b) requires that the licensee use, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to the public that are ALARA. State that procedures will be written and implemented for the ALARA program and that interfaces will be established throughout the activities associated with the geologic repository operations area to meet ALARA requirements. Identify health physics plans, procedures, and mode of operation for ensuring that occupational radiation doses will be ALARA during facility operations, especially for work performed in radiation areas; and that residual contamination levels are ALARA for all systems that contain,

collect, store, or transport radioactive solids and liquids, including those from the radioactive waste treatment, handling, and storage systems.

Characterize any programs, manuals, or other management plans, such as the operations ALARA manual, the radiation protection manual, or the offsite calculation dose manual that will be developed to implement policies and programs described in this chapter, with a brief overview of the contents of each.

Identify procedures that will be in place to address activities discussed in Regulatory Guide 8.8, Regulatory Position 3 (NRC 1978). Include a description of procedures that will address the use of operational data and trend analyses to help reduce radiation doses.

10 CFR 60.21(c)(7) requires that the Safety Analysis Report include a description of the program for control and monitoring of occupational radiation doses to maintain such doses in accordance with the requirements of 10 CFR 20. Describe the program for monitoring and controlling personnel dose, including the bases for that program.

Describe the bases and methods for controlling access to areas identified in the geologic repository operations area that are subject to radiation monitoring. Describe the physical and administrative measures for controlling access and stay time for designated radiation areas. Include a brief discussion of the work controls for ensuring proper protective equipment is worn and proper monitoring is provided.

10 CFR 20.1601 establishes specific requirements for control of access to high radiation areas. Discuss the methods to be employed for control of access to these areas. Specifically address each of the provisions of 10 CFR 20.1601(a) through (e). 10 CFR 20.1602 requires additional measures to be taken for control of access to very high radiation areas. Discuss the additional measures that will be used to ensure that an individual is not able to gain unauthorized or inadvertent access to very high radiation areas. Review Regulatory Guide 8.38, *Control of Access to High and Very High Radiation Areas in Nuclear Power Plants* (NRC 1993a). Some of the guidance is potentially useful and applicable. The project will assess the applicability of the guidance in these documents, and appropriate instructions for the License Application author will be provided in a future revision of the technical guidance document.

Address reduction in radiation dose through use of temporary shielding and through equipment decontamination.

10 CFR 20.1502 identifies conditions requiring individual monitoring of external and internal occupational dose. 10 CFR 20.2104 specifies requirements regarding determination of prior occupational dose. Describe the methods and plans for personnel monitoring, including obtaining information regarding prior occupational dose. Review Regulatory Guide 8.34, *Monitoring Criteria and Methods to Calculate Occupational Radiation Doses* (NRC 1992c), and ANSI/HPS N13.30, *Performance Criteria for Radiobioassay* (ANSI/HPS 1996). Some of the guidance is potentially useful and applicable. The project will assess the applicability of the guidance in these documents, and appropriate instructions for the License Application author will be provided in a future revision of the technical guidance document. Describe the dosimetry program and address the potentially applicable guidance provided in ANSI/HPS N13.30

(ANSI/HPS 1996). Describe the methods and plans for personnel monitoring and dosimetry. Include in the description methods for recording and reporting dosimetry results, criteria for performing routine and non-routine bioassays, and accreditation criteria for dosimeter processing. State that the program will participate in NRC approved performance testing programs. Address conformance with the 10 CFR 20.1501(c) requirement that personnel dosimeters be processed and evaluated under a program that holds current personnel dosimetry accreditation from the National Voluntary Laboratory Accreditation Program and is approved for the type of radiation(s) for which the individual wearing the dosimeter is monitored.

Discuss the provisions for monitoring of extremities. Include mention of the radiation work permit system to ensure that proper monitoring equipment is used and that extra coverage by trained radiation protection personnel is available when required.

10 CFR 20.1204 establishes regulations for determination of internal exposure, including related monitoring requirements. Describe the program for internal exposure control and monitoring that will implement these regulations. Regulatory Guide 8.32, *Criteria for Establishing a Tritium Bioassay Program* (NRC 1988), and Regulatory Guide 8.34, *Monitoring Criteria and Methods to Calculate Occupational Radiation Doses* (NRC 1992c), should be considered for potential usefulness and applicability to internal exposure control and monitoring. The project will assess the applicability of the guidance in these documents and appropriate instructions for the License Application author will be provided in a future revision of the technical guidance document.

10 CFR 20.1208 requires that each licensee ensure that the dose to an embryo/fetus during the entire pregnancy, from occupational exposure of a declared pregnant woman, does not exceed 0.5 rem. Describe how the program will implement this requirement. Review Regulatory Guide 8.36, *Radiation Dose to the Embryo/Fetus* (NRC 1992e), and Regulatory Guide 8.13, *Instruction Concerning Prenatal Radiation Exposure* (NRC 1987). Some of the guidance in these documents is potentially useful and applicable. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of the technical guidance document.

10 CFR 20, Subpart H, addresses controls to restrict internal exposure in restricted areas. State that process and engineering controls are the preferred method of controlling concentrations of contamination in air (see 10 CFR 20.1701). Describe the methods and procedures for monitoring, evaluating, and controlling or minimizing potential airborne radioactive material concentrations, including any requirements for special air sampling. Review Regulatory Guide 8.25, *Air Sampling in the Workplace* (NRC 1992a). Some of the guidance is potentially useful and applicable. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of the technical guidance document.

10 CFR 60.21(c)(15)(v) requires that the License Application include plans for conduct of normal activities, including maintenance, surveillance, and testing. Describe the radiation monitoring and survey program in this section and cross-reference to other chapters to prevent duplication of information. Describe the bases and methods for monitoring and controlling or minimizing personnel, equipment, and surface contamination. Discuss in general terms the work

controls in areas with loose surface or airborne radioactivity, including the method for ensuring that proper protective equipment is worn and that proper monitoring is provided.

State that, when it is not practical to apply process or other engineering controls to control the concentrations of radioactive material in air to values below those that define an airborne radioactivity area, monitoring will be increased and other means will be undertaken to limit intakes by one or more of the following means: control of access, limitation of exposure times, use of respiratory protection equipment, or other controls (10 CFR 20.1702).

10 CFR 20.1703 establishes regulations dealing with the use of individual respiratory protection equipment. Commit to implementation of a respiratory protection program as addressed in these regulations, and describe the program addressing each of the elements of the regulation or referencing applicable discussions that occur elsewhere in this chapter. Review NUREG-0041, *Manual of Respiratory Protection Against Airborne Radioactive Materials* (NRC 1976b), and Regulatory Guide 8.15, *Acceptable Programs for Respiratory Protection* (NRC 1976a). Some of the guidance in these documents is potentially useful and applicable. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of the technical guidance document. Broadly, describe the following elements of the respiratory protection program:

- Techniques to be provided and measures taken to ensure that management policy is carried out.
- Selection, fitting, use, and maintenance of respiratory protective equipment.
- Written procedures to address specified elements of the program.
- Air sampling and other surveys to support the program.
- Bioassays related to the respiratory protection program. Ensure consistency with the commitment to Regulatory Guide 8.9 [NRC 1993b], Section 10.4, Dose Assessment.
- Use of unapproved respiratory protection equipment or assignment of protection factors in excess of those given in Regulatory Guide 8.15, Table I (NRC 1976a).

If respiratory protection equipment is used, 10 CFR 20.1703(a)(3)(ii) requires surveys and bioassays, as appropriate, to evaluate actual intakes. Discuss the bioassay program as it generally applies to radiation protection, including the criteria for performing bioassays. Review ANSI/HPS N13.30, *Performance Criteria for Radiobioassay* (ANSI/HPS 1996). Some of the guidance in this standard is potentially useful and applicable. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of the technical guidance document.

Discuss the criteria for performing routine whole body and lung counting and nonroutine bioassays associated with specific exposures. Indicate how the program follows the guidance provided by Regulatory Guide 8.9, *Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program* (NRC 1993b), and Regulatory Guide 8.2, *Guide For Administrative*

Practices in Radiation Monitoring (NRC 1973). Some of the guidance in these regulatory guides is potentially useful and applicable. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of the technical guidance document.

10 CFR 20.2106 and 10 CFR 20.2203 identify records and reports required to be maintained associated with individual monitoring, exposures, radiation levels, and concentrations. Commit to maintenance of the records and reports associated with facility contamination surveys, internal and external exposure, and respiratory protection program and personnel survey results. Review Regulatory Guide 8.7, *Instructions for Recording and Reporting Occupational Radiation Exposure Data* (NRC 1992b). Some of the guidance in this regulatory guide is potentially useful and applicable. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of the technical guidance document.

Describe how personnel dose status will be disseminated. Describe how dosimetric results are used as a guide to operational planning. Describe methods for recording and reporting results, both on a short-term and on a less frequent basis, such as notices to workers when dose limits are approached, and providing workers with dose records upon departure. Describe how dose verification will occur as established dose limits are approached and discuss dose verification action levels that will be in place to ensure that personnel do not exceed established regulatory and administrative limits. Discuss the controls for authorization to exceed limits during special or emergency situations. Cross-reference to other sections of this chapter and other License Application chapters, as appropriate.

10 CFR 19.11 addresses posting of notices to workers. State that the regulations, notices, license, procedures, and NRC Form 3 required by this regulation are posted as required or the locations are posted where the documents may be located, as addressed in 10 CFR 19.11(b), for the periods required, and in locations pursuant to the regulation. In addition, 10 CFR 19.12(a)(1) requires that workers who are likely to receive occupational dose in excess of 100 mrem be kept informed of the storage, transfer, or use of radiation and/or radioactive material. 10 CFR 19.12(a)(6) identifies requirements regarding advising workers of the radiation dose reports that they may request. Briefly, describe how this will be accomplished.

State that records of planned special exposures will be maintained in accordance with 10 CFR 20.2105 to be addressed in this section and in Section 11.4 of the License Application. Review Regulatory Guide 8.35, *Planned Special Exposures* (NRC 1992d), for applicability to demonstrate compliance with the requirements in 10 CFR 20.1206 on planned special exposures. Some of the guidance in this regulatory guide is potentially useful and applicable. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of the technical guidance document.

10 CFR 20.1501(a) requires that licensees make or cause to be made surveys that may be necessary for the licensee to comply with regulations and are reasonable, under the circumstances, to evaluate the extent of radiation levels, concentrations or quantities of radioactive material, and the potential radiological hazards that could be present. In broad,

generalized terms describe the methods, frequencies, and plans for conducting radiation surveys, area sampling, and air sampling to meet these regulations. Describe the aspects of the program to ensure that surveys are adequate and timely. Include information for ensuring that the survey results will be disseminated in a timely manner. Review Regulatory Guide 8.21, *Health Physics Surveys for Byproduct Material at NRC-Licensed Processing and Manufacturing Plants* (NRC 1979a). Some of the guidance is potentially useful and applicable. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of the technical guidance document.

Describe the actions to be taken for ensuring that occupational radiation doses received by personnel during these surveys/surveillances will be ALARA. Describe the procedures that will be developed to control and perform survey/surveillance activities, including specification of methods, frequencies, locations, and sampling methods. State that records of surveys/surveillances will be retained as required by 10 CFR 20.2103, as addressed in Chapter 11 of the License Application.

Cross-reference Section 11.8 for a description of the training program. State that minimum levels of training and qualifications for management, supervisors, and employees on ALARA principles and worker protection will be established as required by 10 CFR 19.12.

Cross-reference Section 11.4 for a description of the program for management of records and for reporting requirements addressed in 10 CFR 20, Subpart L, Records.

10 CFR 20.1101(c) requires that the licensee periodically (at least annually) review the radiation protection program content and implementation. 10 CFR 20.2102(a)(2) requires that records be retained of audits and other reviews of the content and implementation of the radiation protection program. Regulatory Guide 8.10, Section C.1.b, *Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable* (NRC 1977), identifies that management should periodically perform a formal audit to determine how doses might be lowered. State that audits will include evaluation of the ALARA program and its effectiveness in dose reduction.

Address the methods to be used to implement these regulations or commitments. If it is the intent to apply the audit and surveillance process of the quality assurance program to the radiation protection program requirements for verification of program, compliance, and effectiveness in accordance with the commitments of the License Application and implementing procedures, then state that intent. If other methods, or a combination of methods, of meeting the requirements will be exercised, then briefly describe those methods and address the criteria which will be used for the evaluation.

10.1.3 Low-Level Radioactive Waste Management

Describe in broad terms the approach to low-level radioactive waste management that will be used at the facility, such as what is intended for release versus what is intended for offsite shipment. Address the methods to be used for disposal of airborne, liquid, and solid waste forms. Refer to 10 CFR 20, Subpart K, for information for this section. Additionally, 10 CFR

60.21(c)(15)(vi) requires that the License Application include plans for the decontamination or dismantlement of surface facilities. Include the following in this description:

- Provisions for waste storage (cross-reference Chapter 4, as applicable)
- Method of ensuring waste minimization (cross-reference Chapter 9, as applicable)
- Provisions for safe equipment and facility decontamination (cross-reference Chapter 4, as applicable)
- Method of disposing of ventilation filters (cross-reference Chapter 9, as applicable)
- Process for characterization of waste to satisfy disposal facility waste acceptance criteria (cross-reference Chapter 9, as applicable)
- Provision for compliance with U.S. Department of Transportation requirements for shipping
- Disposal of low-level radioactive waste offsite (cross-reference Chapter 9, as applicable)

10.2 AS LOW AS REASONABLY ACHIEVABLE DESIGN FEATURES

Describe the primary considerations used during the design or modification of facilities or systems to maintain doses ALARA. Describe the optimization methods and design objectives for maintaining personnel doses ALARA. Specific details pertaining to radiation protection design features of individual structures, systems, and components are contained in Chapters 2, 4, 9, and 12. The author of this section should pursue a coherent approach to ALARA across the various design functions to ensure consistency between the information presented in this section and the detailed information discussed in other chapters in the License Application.

10 CFR 60.21(c)(3)(i) requires that the Safety Analysis Report include a demonstration that the requirements of 10 CFR 60.111(a) are met for Category 1 design basis events. 10 CFR 60.111(a) requires that the geologic repository operations area be designed so that until permanent closure has been completed, radiation exposures, radiation levels, and releases of radioactive materials to unrestricted areas will be maintained within the limits specified in 10 CFR 20 and such generally applicable environmental standards for radioactivity as may have been established by the U.S. Environmental Protection Agency. 10 CFR 60.131(a) requires that the geologic repository operations area be designed to maintain radiation doses, levels, and concentrations of radioactive material in air in restricted areas within the limits specified in 10 CFR 20. 10 CFR 20.1101 requires that the licensee use, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are ALARA. Describe implementation of guidance contained in Regulatory Guide 8.8, Position 2 (NRC 1978), to achieve ALARA in design and engineering controls.

Provide evidence that ALARA is an integral factor in the design process of the facility and its systems, including a discussion of the radiological safety features in the designs for processing, transporting, handling, storage, retrieval, emplacement, isolation of radioactive waste, and performance confirmation. Include the layouts of the facilities for radiation protection, including locations of radiological monitoring instruments, and the radiological design features of each system (refer to Chapters 4 and 9 for this information). Cost-benefit analyses are considered by NRC to be an integral part of a viable ALARA program. Lack of cost-benefit analyses related to ALARA design features are considered to be de facto evidence that the ALARA program is not properly implemented. Discuss, or refer to discussions in other chapters, examples of cost-benefit analysis associated with the design.

The discussion is expected to be broad in character but must accurately reflect design. In cases in which actual design information does not exist, discuss the concepts that will be incorporated into the design. In many cases, the design information will have been discussed in detail in prior sections of the License Application. This section may refer to those sections. However, the ALARA features of the design should be detailed in this section.

10.2.1 Facility Design Features

Referring to the radiation sources identified in Section 10.4.1 and the detailed descriptions in other chapters, as appropriate, describe how the radiological designs of the surface and underground physical facilities in the geologic repository operations area will permit safe handling and storage of spent fuel, high-level radioactive waste, and site-generated low-level radioactive wastes during operations and retrieval. Include in the discussion how design will help to limit the time required to perform work in the vicinity of radioactive materials, including, as appropriate, designing equipment for ease of repair and replacement and providing adequate space for ease of operation (10 CFR 60.131(a)(2)).

Provide layout drawings that show controlled-access areas, personnel and equipment decontamination areas, and contamination control areas with type of controls, traffic patterns, location of health physics facilities, location of control panels for radwaste equipment and components, location of onsite laboratories for analysis of chemical and radioactive samples, and location of the counting room. Describe the facilities and equipment shown on the drawings, including any special equipment provided specifically for radiation protection. Describe how relevant experience from any past designs was, and will be, used to develop improved designs for ensuring that occupational doses will be ALARA and that contamination will be minimized. Include a discussion of the radiological safety features in the designs for processing, transporting, handling, storage, retrieval, emplacement, and isolation of radioactive waste.

Discuss the elements of facility design that provide a means to control access to high radiation areas and to high airborne radioactivity areas (10 CFR 60.131(a)(5)). For the discussion on access control to high radiation areas, include a description of the design elements to be incorporated that address the requirements of 10 CFR 20.1601(a) through (d), specifically:

- Control devices meeting the requirements of 10 CFR 20.1601(a)(1) and/or (2)
- Locked entryways
- Continuous direct or electronic surveillance, if used for control of access

- Alternative methods proposed for controlling access, if any
- Means to allow leaving the area

Discuss design elements for control of access to very high radiation areas that are in addition to the requirements for control of access to high radiation areas as specified in 10 CFR 20.1602.

Describe how the facility layout and equipment design contributes to ALARA goals. Include in the description a discussion of the radiological safety philosophy incorporated into the designs for processing, transporting, handling, storage, retrieval, emplacement, and isolation of radioactive waste.

Describe how the design will be directed toward reducing the need for maintenance of equipment, reducing radiation levels and time spent where maintenance is required, and controlling contamination in handling, transfer, and storage of all radioactive materials.

10.2.2 Shielding

10 CFR 60.131(a)(3) requires that the geologic repository operations area design include suitable shielding to control doses within the limits of 10 CFR 20. Referencing the sources identified in Section 10.4.1 as the bases for the radiological designs for shielding, describe in broad terms how the geologic repository operations area design will ensure adequate shielding for dose control during waste package handling, storage, and routine operations of the facility, and provide information on the shielding for each of the radiation sources identified. Develop the context for the assumptions, including cost per person-rem evaluations for design, and sources of radiation used as the bases for the radiological shielding designs. Include a discussion of shielding anticipated to be required for the spent fuel and vitrified high-level radioactive waste sources. Identify shielding anticipated to be required for equipment other than the spent fuel, such as tanks, waste handling equipment, and ventilation filter trains. Describe special protective features to ensure that occupational doses to radiation will be ALARA in normally occupied areas, such as features to support the use of portable shielding, design considerations that provide distance between serviceable components (geometric arrangement), or remote handling equipment. Include layout drawings that show shield wall thicknesses and anticipated doses.

Show the location of each source identified in Section 10.4.1 layout and arrangement drawings. Include the designation of each restricted area, including the boundary of the area and the type of interface with adjacent areas, such as partitions, locked doors, or barriers. Provide information on the shielding for each of the sources. Discuss the general design of penetrations as the design relates to shielding, including the method by which the shield parameters, such as attenuation coefficients and buildup factors were determined and the assumptions, codes, and techniques used in the calculations. Reference other sections of the License Application as appropriate, but discuss ALARA features in detail.

10.2.3 Ventilation

10 CFR 60.132(b) requires that the surface facility ventilation systems supporting waste transfer, inspection, decontamination, processing, packaging, or performance confirmation be designed to provide protection against radiation exposures and offsite releases. Describe how the surface

ventilation system achieves this requirement. Describe the geologic repository operations area ventilation systems as a means to limit concentrations of radioactive material in air (10 CFR 60.131(a)(1)), including how the ventilation system functions as a means to control the dispersal of radioactive contamination (10 CFR 60.131(a)(4)). Discuss how the air treatment systems maintain releases ALARA during normal operations and confine radioactive material during expected operating conditions. This discussion is not intended to duplicate the detailed design information in Chapter 4 on the surface and underground ventilation systems, but rather to focus on how the features of the systems contribute to radiation protection and ALARA goals.

Provide a general level discussion of the purposes of the surface and underground facility ventilation systems. Identify the subsystems, components, and features that will be in place for radiation protection. Include a discussion of design applications to keep concentrations of radioactive material in air to levels well below the derived air concentration values of 10 CFR 20, Appendix B, Table 1. Describe features to be incorporated into the design of the repository ventilation systems that reflect the following objectives and maintain airborne radioactivity levels and releases ALARA during normal operations:

- Interface with other systems to control the transport of radioactive particulates and gases within and control releases from the underground facility.
- Contain radioactive material that has been deposited, collected, or transported by the system, including a discussion on how the spread of radioactive materials between operating areas and within the ventilation systems will be controlled, including a description of the various radiation zones to be used to limit exposures and contamination.
- Minimize exposure during maintenance, tests, or inspections.
- Reduce high airborne levels when they occur.
- Provide sufficient capacity to reduce airborne concentrations and to confine radioactive material.
- Provide ways to monitor ventilation system performance.
- Separate the ventilation of excavation and waste emplacement areas (10 CFR 60.133(g)(3)).
- Ensure continued functioning during normal operations and under accident conditions.
- Provide for localized control of airborne contamination using auxiliary or temporary ventilation systems.

10.2.4 Radiological Monitoring Instrumentation

10 CFR 60.131(a)(4) requires that the geologic repository operations areas design include a means to monitor the dispersal of radioactive contamination in order to limit doses, levels, and concentrations of radioactive materials in the restricted area. 10 CFR 60.131(a)(6) requires that the geologic repository operations area's design include a radiation alarm system to warn of significant increases in radiation levels and in concentrations of radioactive material in air, including provisions for calibration and for testing the alarm system operability. Discuss how the radiation monitoring instrumentation and alarm system meet these requirements.

Describe radiation monitoring and engineering philosophies used for fixed-area radiation monitors and continuous airborne-monitoring instruments. If the area radiation and airborne radioactivity monitoring instrumentation has been described in Chapter 4, refer to the applicable sections and do not repeat the information here. Discuss the fixed-area radiation monitoring instrumentation, the continuous airborne-monitoring instrumentation, and the alarm system to be provided in the geologic repository operations area and how it will meet the requirements identified above. Provide information on auxiliary and emergency power supplies, range, sensitivity, accuracy, energy-dependence, recording devices, and location of detectors, readouts, and alarms. Relate design and operational considerations to ALARA concepts. Regulatory Guide 8.8, Position 2(g) (NRC 1978), provides guidance for central or "built-in" monitoring systems in selected station areas as a method for dose reduction. Address design concepts, including the following:

- Readout capability at an appropriate control point
- Placement and types of detectors for optimum coverage of areas (show on scaled drawings)
- Circuitry that indicates component failure
- Provisions for calibration and testing, including frequencies
- Local alarm readout
- Clear and unambiguous readout
- Ranges adequate to ensure readout of highest anticipated radiation levels and to ensure positive readout at the lowest anticipated levels
- Recording capability

Describe in general terms the concepts involved in placement of monitors and criteria for determining alarm set points. Provide the location of the continuous airborne-monitoring sample collectors, and give details of sampling-line pump location.

10.3 RADIATION PROTECTION OF THE PUBLIC AND ENVIRONMENT

Describe actions taken for radiation protection of the public and the environment. Summarize, without repeating related information provided in Chapters 7, 8, and 9 pertaining to radiation exposures to members of the public. Describe how the ALARA objectives will be set, measured, reviewed, and relayed to the public.

10 CFR 60.21(c)(3)(i) requires that the Safety Analysis Report include a demonstration that the requirements of 60.111(a) are met for Category 1 design basis events. 10 CFR 60.111(a) requires that, until permanent closure, radiation exposures, radiation levels, and releases of radioactive materials to unrestricted areas be maintained at all times within the limits of 10 CFR 20 and any generally applicable environmental standards established by the U. S. Environmental Protection Agency. 10 CFR 20.1302(a) states that the licensee shall make or cause to be made, as appropriate, surveys of radiation levels in unrestricted and controlled areas[5] to demonstrate compliance with the dose limits for individual members of the public in 10 CFR 20.1301. Describe the environmental radiological monitoring and surveillance program that, when used in conjunction with mathematical models, will show compliance with the performance objectives in 10 CFR 60.111(a) and 10 CFR 20.1302(a), and will provide reasonable assurance that the engineered systems, structures, and components in the geologic repository operations area are functioning as intended and as anticipated. Also, describe any interface with the meteorological monitoring program as necessary to complement the radiological monitoring results for confirming the validity of estimates of environmental radiation exposures and doses to the public.

In general terms, describe the approach that will be employed for systematic monitoring of the environment. Provide information identifying anticipated pathways that lead to the highest potential external and internal radiation exposures of individuals resulting from operations. Discuss the details of the approach and the surveys to be performed in unrestricted and controlled areas and the samples to be taken to meet the CFR sections identified above. Refer to Section 10.1.2(e) regarding the survey practices to be used. Describe the process that will be undertaken during the construction period to determine the preoperational background levels, including the time period over which associated data will be collected. Review Regulatory Guide 4.1, *Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants* (NRC 1975). This guide provides potentially useful and applicable information regarding the preoperational program, operational program, and detection capabilities. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of the technical guidance document.

10 CFR 20.1302(b) provides alternative methods for demonstrating compliance with the annual dose limits to individual members of the public contained in 10 CFR 20.1301. Identify the methodologies that will be used, and discuss the processes for translating the results of the monitoring program to estimates of dose contribution of the geologic repository operations area

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[5] 10 CFR 20.1302(a) also requires surveys of radioactive materials in effluents released to unrestricted and controlled areas. That part of the regulation is addressed in Chapter 9 of the technical guidance document.

to the environment. If an independent document, such as an offsite dose calculation manual, is to be created that identifies methodologies for identifying required environmental surveys and for translating the results of those surveys into dose equivalents, referencing that document is acceptable.

10 CFR 20.1101(d) requires a special ALARA consideration be established for a constraint on air emissions of radioactive material to the environment, excluding Radon-222 and its daughters. The constraint is that the individual member of the public likely to receive the highest dose will not be expected to receive a total effective dose equivalent to in excess of 10 mrem (0.1 mSv) per year from these emissions. Discuss the actions necessary to ensure compliance with this regulation. Include the reference to the analyses and calculations performed to support substantiation of compliance, and include monitoring activities that will be performed to ensure evidence of compliance. State that prompt corrective action will be taken if constraints are exceeded and that reports related to exceeding the constraint will be made as stated in Chapter 11, as addressed in 10 CFR 20.2303(a)(2)(vi). Review Regulatory Guide 4.20, *Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees Other Than Power Reactors* (NRC 1996a), and Regulatory Guide 8.37, *ALARA Levels for Effluents From Materials Facilities* (NRC 1993c). Some of the guidance in these regulatory guides is potentially useful and applicable. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of the technical guidance document. References to other chapters of the License Application may be used to support discussion of ALARA levels for gaseous and liquid effluents, if the referenced section contains the information necessary to provide the required evidence of meeting the regulation. As an example, effluent monitoring is discussed in Chapter 9, and that discussion will provide information for monitoring air emissions to the environment and related doses to members of the public.

Describe in general terms any interface that will exist between the program for meteorological monitoring and data collection discussed in Chapter 2 of the License Application and the environmental monitoring program. Describe in general terms how the data from the meteorological program supports assessment of the potential dispersion of radioactive materials and potential radiological consequences to the public and the environment. State that changes to the radiological environmental monitoring program may occur as a result of changes identified through the meteorological monitoring process.

State that procedures will be developed to control the environmental monitoring program, including specification of methods, frequencies, locations, and sampling methods. Review Regulatory Guide 4.15, *Quality Assurance for Radiological Monitoring Programs (Normal Operations) Effluent Streams and the Environment* (NRC 1979b). Some of the guidance is potentially useful and applicable. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of the technical guidance document. References to other chapters of the License Application may be used to support this discussion if the referenced section contains the information necessary to provide the required evidence of meeting the regulation.

10.4 DOSE ASSESSMENT

Provide a description of sources of radiation associated with operations of the MGR and dose assessments for the occupational exposures expected to be received from these sources in the operation of the facility.

10.4.1 Radiation Sources

Identify the sources and levels of radiation within and external to the facility that will present potential hazards to the facility workers. This information must be considered in detail, since layout and shielding design, location of radiation areas, potential need for atmospheric cleanup, and other considerations depend on this advance information.

NOTE: Information on these identified sources will be used to address the dose assessment discussed in Section 10.4.2.

Describe the sources and levels of radiation within the underground and surface facilities as well as within any other areas of the site boundary. Include a statement of the kinds, such as isotopic composition, and quantities of radioactive waste to be received, possessed, stored, and disposed of in the geologic repository operations area (10 CFR 60.31(a)(1)(ii) and 10 CFR 60.101(b)). Cross-reference other chapters of the License Application for detailed discussion of source information. Identify the sources that were used as the bases for shielding design. For x-ray, gamma ray, and neutron sources, include the energy groupings considered, neutron and photon yields, and the source geometries. In addition to the spent fuel and vitrified waste sources, describe sources of radiation, such as tanks that store radioactive fluids, ventilation filter trains that collect airborne radioactive particulate and gases, piping systems, and deposits that may contain radioactive fluids. Identify those that are expected to require shielding. Source terms associated with the radioactive waste management systems are described in Chapter 9 of the License Application and may be referenced here to avoid repetition.

Show the location of each source on layout and arrangement drawings. Include the designation of each restricted area, including the boundary of the area and the type of interface with adjacent areas, such as partitions, locked doors, or barriers.

In addition to the sources of direct radiation, discuss the sources and levels of anticipated airborne radiation. Include information regarding the origin of these sources, such as leakage and evaporation. Include the expected release and release rates of low-level airborne radioactivity from the geologic repository operations area (see Chapter 9) as a source of airborne radiation within the site and its facilities. This influence depends on such factors as levels of radioactivity released, meteorological conditions that cause the activity to linger in the vicinity, and the design of ventilation systems that allows the radioactivity to be drawn into site buildings and facilities.

10.4.2 Dose Assessment

10 CFR 60.21(c)(15)(v) requires that the License Application include plans for conduct of normal activities, including maintenance, surveillance, and testing. 10 CFR 60.131(a) requires

that the geologic repository operations area be designed to maintain radiation doses within the limits specified in 10 CFR 20 (specifically, Subpart C, Occupational Dose Limits (10 CFR 20.1201 through 1208)). To demonstrate the ability to comply with these requirements, report the results of a dose assessment, which is based on the radiation sources identified in Section 10.4.1, and on the radiation protection design features discussed in Section 10.2. Include the source terms associated with the radwaste management systems, identified in Chapter 9, in the bases for the dose assessment.[6] The dose assessment is an estimate of the annual radiation dose to personnel at the geologic repository operations area during normal operations. The purpose of the requirement to perform a dose assessment is to ensure that adequate attention is given to applicable ALARA considerations (NRC 1978).

Categorize anticipated geologic repository operations area activities and develop dose estimates for workers performing these activities. Examples of activity categories include but are not limited to waste handling in the surface facility, waste emplacement, routine maintenance, test and inspection, and special maintenance. Also provide the objectives and criteria for design dose rates in the areas identified.

For areas with expected airborne radioactivity concentrations during normal operation and anticipated operational occurrences, provide estimated man-hours of occupancy and estimated inhalation doses to personnel.

Tabulate the projected annual radiation doses to workers in the various categories during normal operations and anticipated operational occurrences and provide a summation of the doses. As the bases for projections, use locations of occupancy, number of personnel, annual duration of occupancies, and anticipated radiation levels in the locations.

Use tables and layout drawings, even if conceptual, to assist in providing this information. Discuss the methods or models used to perform the estimate and provide the bases for the assumptions used for the assessment. Discuss computer models used to assist the assessment and related approaches or methodologies (reference to computer models already available to NRC may be made by summary only). Identify computer programs or simulations used for the analysis. Identify ALARA considerations that result in dose reductions for this assessment, such as use of remote handling equipment, either quantitatively or qualitatively. Compare the results of the assessment to the numerical limits of 10 CFR 20. Discuss uncertainties in the methodologies used, identify conservatism in the assumptions, and reference published data associated with the analysis.

V. REFERENCES

10 CFR 19 (U.S. Code of Federal Regulations) 1996. Notices, Instructions and Reports to Workers: Inspection and Investigations.

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[6]Dose assessments for members of the public are contained in Chapter 7 based on anticipated effluent radioactive release information contained in Chapter 9.

10 CFR 20. Standards for Protection Against Radiation.

10 CFR 60. Disposal of High-Level Radioactive Wastes in Geologic Repositories.

ANSI/HPS (American National Standards Institute/Health Physics Society) 1996. *Performance Criteria for Radiobioassay.* ANSI/HPS N13.30. La Grange Park, Illinois: Health Physics Society. 238098.

NRC (U.S. Nuclear Regulatory Commission) 1973. *Guide for Administrative Practices in Radiation Monitoring.* Regulatory Guide 8.2. Rev. 0. Washington, D.C.: U.S. Government Printing Office. 2972.

NRC 1975. *Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants.* Regulatory Guide 4.1. Rev. 1. Washington, D.C.: U.S. Government Printing Office. 3698.

NRC 1976a. *Acceptable Programs for Respiratory Protection.* Regulatory Guide 8.15. Rev. 0. Washington, D.C.: U.S. Government Printing Office. 3003.

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CHAPTER 11. CONDUCT OF OPERATIONS AND RELATED TOPICS

I. OBJECTIVE

This chapter provides guidance to the author(s) of Chapter 11 of the License Application. The License Application will be submitted to the U.S. Nuclear Regulatory Commission (NRC) for authorization to construct a geologic repository for high-level radioactive waste at Yucca Mountain. The author(s) will need to demonstrate that the U.S. Department of Energy (DOE) has workable concepts for the conduct of repository operations. This chapter will present information describing the conduct of repository operations and the associated procedures, including planned activities and processes, maintenance, inspection and testing, facility organization and personnel, procedure development, records and reports, operating controls and limits, personnel training, emergency planning, material control and accountability, and proposed technical specifications. This chapter demonstrates that DOE will establish and maintain a staff of adequate size and technical competence and that operating plans to be followed are adequate to protect public health and safety.

II. GUIDANCE TOPICS

The following topics will be discussed in Chapter 11 of the License Application:

- 11.0 PURPOSE, SUMMARY, AND CONCLUSIONS
- 11.1 CONDUCT OF OPERATIONS
  - 11.1.1 Repository Construction
  - 11.1.2 Startup Activities and Testing
  - 11.1.3 Waste Handling Activities
  - 11.1.4 Surveillance and Periodic Testing
  - 11.1.5 Maintenance
  - 11.1.6 Waste Retrieval, Removal, and Alternate Storage
- 11.2 ORGANIZATIONAL STRUCTURE, MANAGEMENT, STAFFING, AND ADMINISTRATIVE CONTROLS
  - 11.2.1 U.S. Department of Energy
  - 11.2.2 Management and Operating Contractor
  - 11.2.3 Administrative Controls
- 11.3 PROCEDURES
  - 11.3.1 Procedural Development, Approval, and Change Control
  - 11.3.2 Administrative Procedures
  - 11.3.3 Design and Construction/Operating Procedures
- 11.4 RECORDS AND REPORTS
  - 11.4.1 Records Retention
  - 11.4.2 Record Retention Periods
- 11.5 PROPOSED OPERATING CONTROLS AND LIMITS
- 11.6 PRESERVATION OF RECORDS

11.7 SITE MARKERS
11.7.1 General Site Description
11.7.2 Site Marker Description
11.7.3 Additional Access Control Methods
11.8 TRAINING
11.8.1 Training Organization
11.8.2 Development and Qualification of Training Staff
11.8.3 Personnel Selection and Entry-Level Requirements
11.8.4 Facility Personnel Training
11.8.5 Qualification and Certification Process
11.8.6 Training Records
11.9 EMERGENCY PLANNING
11.10 REQUIREMENTS FOR ADDITIONAL TECHNICAL INFORMATION
11.11 PROPOSED TECHNICAL SPECIFICATIONS

III. REGULATORY REQUIREMENTS

***** IMPORTANT NOTE *****

At the time of development of this document, it appears likely that NRC will issue a new, site-specific regulation for Yucca Mountain in 1999. At the time the new 10 CFR 63 is issued, 10 CFR 60 will be modified to state that it is not applicable to Yucca Mountain. Details about the new regulation are not known. However, it has been determined prudent to develop the working draft License Application in accordance with the best available estimate of the form of the new regulation. The form of the new regulation is not expected to affect the context of the requirements for conduct of operations and related topics. Therefore, the text of the technical guidance document that follows this note has not been revised and the working draft License Application authors are expected to follow the guidance provided below. Any questions should be addressed to Management and Operating Contractor (M&O) Licensing.

***** END OF NOTE *****

Sections of the regulations that are to be addressed fully, or in part, in Chapter 11 of the License Application are reproduced below. Paragraph numbers refer to Title 10 of the U.S. Code of Federal Regulations (CFR). These are verbatim extracts from the regulations, with the exception of 10 CFR 60, Subpart I – Emergency Planning Criteria [RESERVED]. Because the emergency planning criteria have yet to be promulgated, information from emergency planning sections of 10 CFR Parts 30, 50, 70, and 72 is provided. This information should be useful in proceeding with reasonable preparations for emergency planning until applicable federal regulations are established. Appendix A provides a cross-reference between the regulations in 10 CFR 60 and the specific License Application sections that address these regulations. Revisions to regulations are not uncommon. Verify that current versions are used.

10 CFR Part 19, Notices, Instructions and Reports to Workers: Inspection and Investigations

19.11 Posting of notices to workers.

(a) Each licensee shall post current copies of the following documents:

(1) The regulations in this part and in Part 20 of this chapter

(2) The license, license conditions, or documents incorporated into a license by reference, and amendments thereto

(3) The operating procedures applicable to licensed activities

(4) Any notice of violation involving radiological working conditions, proposed imposition of civil penalty, or order issued pursuant to Subpart B of Part 2 of this chapter, and any response from the licensee.

(b) If posting of a document specified in paragraph (a) (1), (2) or (3) of this section is not practicable, the licensee may post a notice which describes the document and states where it may be examined.

(c)(1) Each licensee and each applicant for a specific license shall prominently post NRC Form 3 (Revision dated January 1996), "Notice to Employees.

(2) Copies of NRC Form 3 may be obtained by writing to the Regional Administrator of the appropriate U.S. Nuclear Regulatory Commission Regional Office listed in Appendix D to Part 20 of this chapter or by calling the NRC Information and Records Management Branch at (301) 415-7230.

Note: Copies of NRC Form 3 may be obtained by writing to the Regional Administrator of the appropriate U.S. Nuclear Regulatory Commission Regional Office listed in appendix D to part 20 of this chapter or by contacting the NRC Information and Records Management Branch (telephone no. (301) 415-2730).

(d) Documents, notices, or forms posted pursuant to this section shall appear in a sufficient number of places to permit individuals engaged in licensed activities to observe them on the way to or from any particular licensed activity location to which the document applies, shall be conspicuous, and shall be replaced if defaced or altered.

(e) Commission documents posted pursuant to paragraph (a)(4) of this section shall be posted within 2 working days after receipt of the documents from the Commission; the licensee's response, if any, shall be posted within 2 working days after dispatch by the licensee. Such documents shall remain posted for a minimum of 5 working days or until action correcting the violation has been completed, whichever is later.

19.12 Instruction to workers.

(a) All individuals who in the course of employment are likely to receive in a year an occupational dose in excess of 100 mrem (1 mSv) shall be -

(1) Kept informed of the storage, transfer, or use of radiation and/or radioactive material;

(2) Instructed in the health protection problems associated with exposure to radiation and/or radioactive material, in precautions or procedures to minimize exposure, and in the purposes and functions of protective devices employed;

(3) Instructed in, and required to observe, to the extent within the workers control, the applicable provisions of Commission regulations and licenses for the protection of personnel from exposure to radiation and/or radioactive material;

(4) Instructed of their responsibility to report promptly to the licensee any condition which may lead to or cause a violation of Commission regulations and licenses or unnecessary exposure to radiation and/or radioactive material;

(5) Instructed in the appropriate response to warnings made in the event of an unusual occurrence or malfunction that may involve exposure to radiation and/or radioactive material; and

(6) Advised as to the radiation exposure reports which workers may request pursuant to §19.13.

10 CFR Part 20, Standards for Protection Against Radiation
Subpart L - Records
20.2101 General provisions.

(a) Each licensee shall use the units: curie, rad, rem, including multiples and subdivisions, and shall clearly indicate the units of all quantities on records required by this part.

(b) Not withstanding the requirements of paragraph (a) of this section, when recording information on shipment manifests, as required in Sec. 20.2006(b), information must be recorded in the International System of Units (SI) or in SI and units as specified in paragraph (a) of this section.

(c) The licensee shall make a clear distinction among the quantities entered on the records required by this part (e.g., total effective dose equivalent, shallow-dose equivalent, eye dose equivalent, deep-dose equivalent, committed effective dose equivalent).

20.2102 Records of radiation protection programs.

(a) Each licensee shall maintain records of the radiation protection program, including:

(1) The provisions of the program; and

(2) Audits and other reviews of program content and implementation.

(b) The licensee shall retain the records required by paragraph (a)(1) of this section until the Commission terminates each pertinent license requiring the record. The licensee shall retain the records required by paragraph (a)(2) of this section for 3 years after the record is made.

20.2103 Records of surveys.

(a) Each licensee shall maintain records showing the results of surveys and calibrations required by Secs. 20.1501 and 20.1906(b). The licensee shall retain these records for 3 years after the record is made.

(b) The licensee shall retain each of the following records until the Commission terminates each pertinent license requiring the record:

(1) Records of the results of surveys to determine the dose from external sources and used, in the absence of or in combination with individual monitoring data, in the assessment of individual dose equivalents. This includes those records of results of surveys to determine the dose from external sources and used, in the absence of or in combination with individual monitoring data, in the assessment of individual dose equivalents required under the standards for protection against radiation in effect prior to January 1, 1994; and

(2) Records of the results of measurements and calculations used to determine individual intakes of radioactive material and used in the assessment of internal dose. This includes those records of the results of measurements and calculations used to determine individual intakes of radioactive material and used in the assessment of internal dose required under the standards for protection against radiation in effect prior to January 1, 1994; and

(3) Records showing the results of air sampling, surveys, and bioassays required pursuant to Sec. 20.1703(a)(3) (i) and (ii). This includes those records showing the results of air sampling, surveys, and bioassays required under the standards for protection against radiation in effect prior to January 1, 1994; and

(4) Records of the results of measurements and calculations used to evaluate the release of radioactive effluents to the environment. This includes those records of the results of measurements and calculations used to evaluate the release of radioactive effluents to the environment required under the standards for protection against radiation in effect prior to January 1, 1994.

20.2104 Determination of prior occupational dose.

(a) For each individual who is likely to receive in a year, an occupational dose requiring monitoring pursuant to Sec. 20.1502 the licensee shall--

(1) Determine the occupational radiation dose received during the current year; and

(2) Attempt to obtain the records of cumulative occupational radiation dose.

(b) Prior to permitting an individual to participate in a planned special exposure, the licensee shall determine--

(1) The internal and external doses from all previous planned special exposures; and

(2) All doses in excess of the limits (including doses received during accidents and emergencies) received during the lifetime of the individual.

(c) In complying with the requirements of paragraph (a) of this section, a licensee may--

(1) Accept, as a record of the occupational dose that the individual received during the current year, a written signed statement from the individual, or from the individual's most recent employer for work involving radiation exposure, that discloses the nature and the amount of any occupational dose that the individual may have received during the current year;

(2) Accept, as the record of cumulative radiation dose, an up-to- date NRC Form 4, or equivalent, signed by the individual and countersigned by an appropriate official of the most recent employer for work involving radiation exposure, or the individual's current employer (if the individual is not employed by the licensee); and

(3) Obtain reports of the individual's dose equivalent(s) from the most recent employer for work involving radiation exposure, or the individual's current employer (if the individual is not employed by the licensee) by telephone, telegram, electronic media, or letter. The licensee shall request a written verification of the dose data if the authenticity of the transmitted report cannot be established.

(d) The licensee shall record the exposure history of each individual, as required by paragraph (a) of this section, on NRC Form 4, or other clear and legible record, including all of the information required by NRC Form 4[1] . The form or record must show each period in which the individual received occupational

[1] Licensees are not required to partition historical dose between external dose equivalent(s) and internal committed dose equivalent(s). Further, occupational exposure histories obtained and recorded on NRC Form 4 before January 1, 1994, might not have included effective dose equivalent, but may be used in the absence of specific information on the intake of radionuclides by the individual.

exposure to radiation or radioactive material and must be signed by the individual who received the exposure. For each period for which the licensee obtains reports, the licensee shall use the dose shown in the report in preparing the NRC Form 4. For any period in which the licensee does not obtain a report, the licensee shall place a notation on the NRC Form 4 indicating the periods of time for which data are not available.

(e) If the licensee is unable to obtain a complete record of an individual's current and previously accumulated occupational dose, the licensee shall assume--

(1) In establishing administrative controls under Sec. 20.1201(f) for the current year, that the allowable dose limit for the individual is reduced by 1.25 rems (12.5 mSv) for each quarter for which records were unavailable and the individual was engaged in activities that could have resulted in occupational radiation exposure; and

(2) That the individual is not available for planned special exposures.

(f) The licensee shall retain the records on NRC Form 4 or equivalent until the Commission terminates each pertinent license requiring this record. The licensee shall retain records used in preparing NRC Form 4 for 3 years after the record is made. This includes records required under the standards for protection against radiation in effect prior to January 1, 1994.

20.2105 Records of planned special exposures.

(a) For each use of the provisions of Sec. 20.1206 for planned special exposures, the licensee shall maintain records that describe--

(1) The exceptional circumstances requiring the use of a planned special exposure; and

(2) The name of the management official who authorized the planned special exposure and a copy of the signed authorization; and

(3) What actions were necessary; and

(4) Why the actions were necessary; and

(5) How doses were maintained ALARA; and

(6) What individual and collective doses were expected to result, and the doses actually received in the planned special exposure.

(b) The licensee shall retain the records until the Commission terminates each pertinent license requiring these records.

20.2106 Records of individual monitoring results.

(a) Recordkeeping requirement. Each licensee shall maintain records of doses received by all individuals for whom monitoring

was required pursuant to Sec. 20.1502, and records of doses received during planned special exposures, accidents, and emergency conditions. These records must include, when applicable--

(1) The deep-dose equivalent to the whole body, eye dose equivalent, shallow-dose equivalent to the skin, and shallow-dose equivalent to the extremities; and

(2) The estimated intake or body burden of radionuclides (see Sec. 20.1202); and

(3) The committed effective dose equivalent assigned to the intake or body burden of radionuclides; and

(4) The specific information used to calculate the committed effective dose equivalent pursuant to Sec. 20.1204(c); and

(5) The total effective dose equivalent when required by Sec. 20.1202; and

(6) The total of the deep-dose equivalent and the committed dose to the organ receiving the highest total dose

(b) Recordkeeping frequency. The licensee shall make entries of the records specified in paragraph (a) of this section at least annually.

(c) Recordkeeping format. The licensee shall maintain the records specified in paragraph (a) of this section on NRC Form 5, in accordance with the instructions for NRC Form 5, or in clear and legible records containing all the information required by NRC Form 5.

(d) Privacy protection. The records required under this section should be protected from public disclosure because of their personal privacy nature. These records are protected by most State privacy laws and, when transferred to the NRC, are protected by the Privacy Act of 1974, Public Law 93-579, 5 U.S.C. 552a, and the Commission's regulations in 10 CFR Part 9.

(e) The licensee shall maintain the records of dose to an embryo/ fetus with the records of dose to the declared pregnant woman. The declaration of pregnancy shall also be kept on file, but may be maintained separately from the dose records.

(f) The licensee shall retain the required form or record until the Commission terminates each pertinent license requiring this record. This includes records required under the standards for protection against radiation in effect prior to January 1, 1994.

20.2107 Records of dose to individual members of the public.

(a) Each licensee shall maintain records sufficient to demonstrate compliance with the dose limit for individual members of the public (see Sec. 20.1301).

(b) The licensee shall retain the records required by paragraph (a) of this section until the Commission terminates each pertinent license requiring the record.

20.2108 Records of waste disposal.

(a) Each licensee shall maintain records of the disposal of licensed materials made under Secs. 20.2002, 20.2003, 20.2004, 20.2005, 10 CFR Part 61 and disposal by burial in soil, including burials authorized before January 28, 1981.[3]

(b) The licensee shall retain the records required by paragraph (a) of this section until the Commission terminates each pertinent license requiring the record. Requirements for disposition of these records, prior to license termination, are located in Secs. 30.51, 40.61, 70.51, and 72.80 for activities licensed under these parts.

20.2110 Form of records.

Each record required by this part must be legible throughout the specified retention period. The record may be the original or a reproduced copy or a microform provided that the copy or microform is authenticated by authorized personnel and that the microform is capable of producing a clear copy throughout the required retention period. The record may also be stored in electronic media with the capability for producing legible, accurate, and complete records during the required retention period. Records, such as letters, drawings, and specifications, must include all pertinent information, such as stamps, initials, and signatures. The licensee shall maintain adequate safeguards against tampering with and loss of records.

Subpart M–Reports

20.2201 Reports of theft or loss of licensed material.

(a) Telephone reports. (1) Each licensee shall report by telephone as follows:

(i) Immediately after its occurrence becomes known to the licensee, any lost, stolen, or missing licensed material in an aggregate quantity equal to or greater than 1,000 times the quantity specified in appendix C to part 20 under such circumstances that it appears to the licensee that an exposure could result to persons in unrestricted areas; or

(ii) Within 30 days after the occurrence of any lost, stolen, or missing licensed material becomes known to the licensee, all

[3] A previous Sec. 20.304 permitted burial of small quantities of licensed materials in soil before January 28, 1981, without specific Commission authorization.

licensed material in a quantity greater than 10 times the quantity specified in appendix C to part 20 that is still missing at this time.

(2) Reports must be made as follows:

(i) Licensees having an installed Emergency Notification System shall make the reports to the NRC Operations Center in accordance with Sec. 50.72 of this chapter, and

(ii) All other licensees shall make reports by telephone to the NRC Operations Center (301-951-0550).

(b) Written reports. (1) Each licensee required to make a report under paragraph (a) of this section shall, within 30 days after making the telephone report, make a written report setting forth the following information:

(i) A description of the licensed material involved, including kind, quantity, and chemical and physical form; and

(ii) A description of the circumstances under which the loss or theft occurred; and

(iii) A statement of disposition, or probable disposition, of the licensed material involved; and

(iv) Exposures of individuals to radiation, circumstances under which the exposures occurred, and the possible total effective dose equivalent to persons in unrestricted areas; and

(v) Actions that have been taken, or will be taken, to recover the material; and

(vi) Procedures or measures that have been, or will be, adopted to ensure against a recurrence of the loss or theft of licensed material.

(2) Reports must be made as follows:

(i) For holders of an operating license for a nuclear power plant, the events included in paragraph (b) of this section must be reported in accordance with the procedures described in Sec. 50.73(b), (c), (d), (e), and (g) of this chapter and must include the information required in paragraph (b)(1) of this section, and

(ii) All other licensees shall make reports to the Administrator of the appropriate NRC Regional Office listed in Appendix D to part 20.

(c) A duplicate report is not required under paragraph (b) of this section if the licensee is also required to submit a report pursuant to Secs. 30.55(c), 40.64(c), 50.72, 50.73, 70.52, 73.27(b), 73.67(e)(3)(vi), 73.67(g)(3)(iii), 73.71, or Sec. 150.19(c) of this chapter.

(d) Subsequent to filing the written report, the licensee shall also report any additional substantive information on the loss or theft within 30 days after the licensee learns of such information.

(e) The licensee shall prepare any report filed with the Commission pursuant to this section so that names of individuals

who may have received exposure to radiation are stated in a separate and detachable part of the report.

20.2202 Notification of incidents.

(a) Immediate notification. Notwithstanding any other requirements for notification, each licensee shall immediately report any event involving byproduct, source, or special nuclear material possessed by the licensee that may have caused or threatens to cause any of the following conditions--

(1) An individual to receive--

(i) A total effective dose equivalent of 25 rems (0.25 Sv) or more; or

(ii) An eye dose equivalent of 75 rems (0.75 Sv) or more; or

(iii) A shallow-dose equivalent to the skin or extremities of 250 rads (2.5 Gy) or more; or

(2) The release of radioactive material, inside or outside of a restricted area, so that, had an individual been present for 24 hours, the individual could have received an intake five times the annual limit on intake (the provisions of this paragraph do not apply to locations where personnel are not normally stationed during routine operations, such as hot-cells or process enclosures).

(b) Twenty-four hour notification. Each licensee shall, within 24 hours of discovery of the event, report any event involving loss of control of licensed material possessed by the licensee that may have caused, or threatens to cause, any of the following conditions:

(1) An individual to receive, in a period of 24 hours--

(i) A total effective dose equivalent exceeding 5 rems (0.05 Sv); or

(ii) An eye dose equivalent exceeding 15 rems (0.15 Sv); or

(iii) A shallow-dose equivalent to the skin or extremities exceeding 50 rems (0.5 Sv); or

(2) The release of radioactive material, inside or outside of a restricted area, so that, had an individual been present for 24 hours, the individual could have received an intake in excess of one occupational annual limit on intake (the provisions of this paragraph do not apply to locations where personnel are not normally stationed during routine operations, such as hot-cells or process enclosures).

(c) The licensee shall prepare any report filed with the Commission pursuant to this section so that names of individuals who have received exposure to radiation or radioactive material are stated in a separate and detachable part of the report.

(d) Reports made by licensees in response to the requirements of this section must be made as follows:

(1) Licensees having an installed Emergency Notification System shall make the reports required by paragraphs (a) and (b)

of this section to the NRC Operations Center in accordance with 10 CFR Part 50.72; and

(2) All other licensees shall make the reports required by paragraph (a) and (b) of this section by telephone to the NRC Operations Center (301) 816-5100 and by telegram, mailgram, or facsimile to the Administrator of the appropriate NRC Regional Office listed in appendix D to 10 CFR Part 20.

(e) The provisions of this section do not include doses that result from planned special exposures, that are within the limits for planned special exposures, and that are reported under Sec. 20.2204.

20.2203 Reports of exposures, radiation levels, and concentrations of radioactive material exceeding the constraints or limits.

(a) Reportable events. In addition to the notification required by Sec. 20.2202, each licensee shall submit a written report within 30 days after learning of any of the following occurrences:

(1) Any incident for which notification is required by Sec. 20.2202; or

(2) Doses in excess of any of the following:

(i) The occupational dose limits for adults in Sec. 20.1201; or

(ii) The occupational dose limits for a minor in Sec. 20.1207; or

(iii) The limits for an embryo/fetus of a declared pregnant woman in Sec. 20.1208; or

(iv) The limits for an individual member of the public in Sec. 20.1301; or

(v) Any applicable limit in the license; or

(vi) The ALARA constraints for air emissions established under Sec. 20.1101(d); or

(3) Levels of radiation or concentrations of radioactive material in--

(i) A restricted area in excess of any applicable limit in the license; or

(ii) An unrestricted area in excess of 10 times any applicable limit set forth in this part or in the license (whether or not involving exposure of any individual in excess of the limits in Sec. 20.1301); or

(4) For licensees subject to the provisions of EPA's generally applicable environmental radiation standards in 40 CFR Part 190, levels of radiation or releases of radioactive material in excess of those standards, or of license conditions related to those standards.

(b) Contents of reports. (1) Each report required by paragraph (a) of this section must describe the extent of exposure of

individuals to radiation and radioactive material, including, as appropriate:

(i) Estimates of each individual's dose; and

(ii) The levels of radiation and concentrations of radioactive material involved; and

(iii) The cause of the elevated exposures, dose rates, or concentrations; and

(iv) Corrective steps taken or planned to ensure against a recurrence, including the schedule for achieving conformance with applicable limits, ALARA constraints, generally applicable environmental standards, and associated license conditions.

(2) Each report filed pursuant to paragraph (a) of this section must include for each occupationally overexposed[4] individual: the name, Social Security account number, and date of birth. The report must be prepared so that this information is stated in a separate and detachable part of the report.

(c) For holders of an operating license for a nuclear power plant, the occurrences included in paragraph (a) of this section must be reported in accordance with the procedures described in Sec. 50.73(b), (c), (d), (e), and (g) of this chapter and must also include the information required by paragraph (b) of this section. Occurrences reported in accordance with Sec. 50.73 of this chapter need not be reported by a duplicate report under paragraph (a) of this section.

(d) All licensees, other than those holding an operating license for a nuclear power plant, who make reports under paragraph (a) of this section shall submit the report in writing to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the appropriate NRC Regional Office listed in appendix D to part 20.

20.2204 Reports of planned special exposures.

The licensee shall submit a written report to the Administrator of the appropriate NRC Regional Office listed in appendix D to part 20 within 30 days following any planned special exposure conducted in accordance with Sec. 20.1206, informing the Commission that a planned special exposure was conducted and indicating the date the planned special exposure occurred and the information required by Sec. 20.2105.

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[4] With respect to the limit for the embryo-fetus (Sec. 20.1208), the identifiers should be those of the declared pregnant woman.

20.2205 Reports to individuals of exceeding dose limits.

When a licensee is required, pursuant to the provisions of Secs. 20.2203, 20.2204, or 20.2206, to report to the Commission any exposure of an identified occupationally exposed individual, or an identified member of the public, to radiation or radioactive material, the licensee shall also provide a copy of the report submitted to the Commission to the individual. This report must be transmitted at a time no later than the transmittal to the Commission.

20.2206 Reports of individual monitoring.

(a) This section applies to each person licensed by the Commission to--

(1) Operate a nuclear reactor designed to produce electrical or heat energy pursuant to Sec. 50.21(b) or Sec. 50.22 of this chapter or a testing facility as defined in Sec. 50.2 of this chapter; or

(2) Possess or use byproduct material for purposes of radiography pursuant to Parts 30 and 34 of this chapter; or

(3) Possess or use at any one time, for purposes of fuel processing, fabricating, or reprocessing, special nuclear material in a quantity exceeding 5,000 grams of contained uranium-235, uranium-233, or plutonium, or any combination thereof pursuant to part 70 of this chapter; or

(4) Possess high-level radioactive waste at a geologic repository operations area pursuant to part 60 of this chapter; or

(5) Possess spent fuel in an independent spent fuel storage installation (ISFSI) pursuant to part 72 of this chapter; or

(6) Receive radioactive waste from other persons for disposal under part 61 of this chapter; or

(7) Possess or use at any time, for processing or manufacturing for distribution pursuant to parts 30, 32, 33 or 35 of this chapter, byproduct material in quantities exceeding any one of the following quantities:

(b) Each licensee in a category listed in paragraph (a) of this section shall submit an annual report of the results of individual monitoring carried out by the licensee for each individual for whom monitoring was required by Sec. 20.1502 during that year. The licensee may include additional data for individuals for whom monitoring was provided but not required. The licensee shall use Form NRC 5 or electronic media containing all the information required by Form NRC 5.

Cesium-137	1
Cobalt-60	1
Gold-198.	100
Iodine-131	1

Iridium-192.	10
Krypton-85	1,000
Promethium-147	10
Techetium-99m	1,000

(c) The licensee shall file the report required by Sec. 20.2206(b), covering the preceding year, on or before April 30 of each year. The licensee shall submit the report to the REIRS Project Manager, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

10 CFR Part 30, Rules of General Applicability to Domestic Licensing of Byproduct Material

30.32 Application for specific licenses.

(i)(1) Each application to possess radioactive materials in unsealed form, on foils or plated sources, or sealed in glass in excess of the quantities in § 30.72, "Schedule C—Quantities of Radioactive Materials Requiring Consideration of the Need for an Emergency Plan for Responding to a Release," must contain either:

An evaluation showing that the maximum dose to a person offsite due to a release of radioactive materials would not exceed 1 rem effective dose equivalent or 5 rems to the thyroid; or

An emergency plan for responding to a release of radioactive material.

An emergency plan for responding to a release of radioactive material submitted under paragraph (i)(1)(ii) of this section must include the following information:...

10 CFR Part 50, Domestic Licensing of Production and Utilization Facilities

50.34 Contents of applications; technical information.

Preliminary safety analysis report. Each application for a construction permit shall include a preliminary safety analysis report. The minimum information to be included shall consist of the following:

A discussion of the applicant's preliminary plans for coping with emergencies.

Appendix E sets forth items which shall be included in these plans.

50.47 Emergency plans.

The onsite and, except as provided in paragraph (d) of this section, offsite emergency response plans for nuclear power reactors must meet the following standards:

10 CFR Part 50 Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants

Subsection XVII, Quality Assurance Records; "Sufficient records shall be maintained to furnish evidence of activities affecting quality. The records shall include at least the following; Operating logs and the results of reviews, inspections, tests, audits monitoring of work performance, and materials analysis. The records shall also include closely-related data such as qualifications of personnel, procedures and equipment. Inspections and test records shall, as a minimum, identify the inspector or data recorder, the type of observation, the results, the acceptability, and the action taken in connection with any deficiency noted. Records shall be identifiable and retrievable. Consistent with applicable regulatory requirements, the applicant shall establish requirements concerning record retention, such as duration, location, and assigned responsibility.

10 CFR Part 60, Disposal of High-Level Radioactive Wastes in Geologic Repositories
Subpart A - General Provisions
60.3 Definitions.
Unanticipated processes and events means...(5) relevant records are preserved, and remain accessible, for several hundred years after permanent closure.

60.3 License required.

(a) The DOE shall not receive or possess source, special nuclear, or byproduct material at a geologic repository operations area except as authorized by a license issued by the Commission pursuant to this part.

(b) The DOE shall not commence construction of a geologic repository operations area unless it has filed an application with the Commission and has obtained construction authorization as provided in this part. Failure to comply with this requirement shall be grounds for denial of a license.

60.4 Communications and records.

(b) Each record required by this part must be legible throughout the retention period specified by each Commission regulation. The record may be the original or a reproduced copy or a microform provided that the copy or microform is authenticated by authorized personnel and that the microform is capable of producing a clear copy throughout the required retention period. The record may also be stored in electronic media with the capability for producing legible, accurate, and complete records during the required retention period. Records such as letters,

drawings, specifications, must include all pertinent information such as stamps, initials, and signatures. The licensee shall maintain adequate safeguards against tampering with and loss of records.

Subpart B - Licenses
License Applications
60.9 Employee protection.

(a) Discrimination by a Commission licensee, an applicant for a Commission license, or a contractor or subcontractor of a Commission licensee or applicant against an employee for engaging in certain protected activities is prohibited. Discrimination includes discharge and other actions that relate to compensation, terms, conditions, or privileges of employment. The protected activities are established in Section 211 of the Energy Reorganization Act of 1974, as amended, and in general are related to the administration or enforcement of a requirement imposed under the Atomic Energy Act or the Energy Reorganization Act.

(1) The protected activities include but are not limited to:

(i) Providing the Commission or his or her employer information about alleged violations of either of the statutes named in paragraph (a) introductory text of this section or possible violations of requirements imposed under either of those statutes;

(ii) Refusing to engage in any practice made unlawful under either of the statutes named in paragraph (a) introductory text or under these requirements if the employee has identified the alleged illegality to the employer;

(iii) Requesting the Commission to institute action against his or her employer for the administration or enforcement of these requirements;

(iv) Testifying in any Commission proceeding, or before Congress, or at any Federal or State proceeding regarding any provision (or proposed provision) of either of the statutes named in paragraph (a) introductory text.

(v) Assisting or participating in, or is about to assist or participate in, these activities.

(2) These activities are protected even if no formal proceeding is actually initiated as a result of the employee assistance or participation.

(3) This section has no application to any employee alleging discrimination prohibited by this section who, acting without direction from his or her employer (or the employer's agent), deliberately causes a violation of any requirement of the Energy Reorganization Act of 1974, as amended, or the Atomic Energy Act of 1954, as amended.

(b) Any employee who believes that he or she has been discharged or otherwise discriminated against by any person for engaging in protected activities specified in paragraph (a)(1) of this section may seek a remedy for the discharge or discrimination through an administrative proceeding in the Department of Labor. The administrative proceeding must be initiated within 180 days after an alleged violation occurs. The employee may do this by filing a complaint alleging the violation with the Department of Labor, Employment Standards Administration, Wage and Hour Division. The Department of Labor may order reinstatement, back pay, and compensatory damages.

(c) A violation of paragraph (a), (e), or (f) of this section by a Commission licensee, an applicant for a Commission license, or a contractor or subcontractor of a Commission licensee or applicant may be grounds for -

(1) Denial, revocation, or suspension of the license.

(2) Imposition of a civil penalty on the licensee or applicant.

(3) Other enforcement action.

(d) Actions taken by an employer, or others, which adversely affect an employee may be predicated upon nondiscriminatory grounds. The prohibition applies when the adverse action occurs because the employee has engaged in protected activities. An employee's engagement in protected activities does not automatically render him or her immune from discharge or discipline for legitimate reasons or from adverse action dictated by nonprohibited considerations.

(e) Each licensee and each applicant for a license shall prominently post the revision of NRC Form 3, "Notice to Employees," referenced in 10 CFR Part 19.11(c). This form must be posted at locations sufficient to permit employees protected by this section to observe a copy on the way to or from their place of work. Premises must be posted not later than 30 days after an application is docketed and remain posted while the application is pending before the Commission, during the term of the license, and for 30 days following license termination.

Note: Copies of NRC Form 3 may be obtained by writing to the Regional Administrator of the appropriate U.S. Nuclear Regulatory Commission Regional Office listed in appendix D to Part 20 of this chapter or by contacting the NRC Information and Records Management Branch (telephone no. (301) 415-7230).

(f) No agreement affecting the compensation, terms, conditions, or privileges of employment, including an agreement to settle a complaint filed by an employee with the Department of Labor pursuant to section 211 of the Energy Reorganization Act of 1974, as amended, may contain any provision which would prohibit, restrict, or otherwise discourage an employee from

participating in protected activity as defined in paragraph (a)(1) of this section including, but not limited to, providing information to the NRC or to his or her employer on potential violations or other matters within NRC regulatory responsibilities. (58 FR 52411, Oct. 8, 1993, as amended at 60 FR 24552, May 9, 1995)

60.17 Contents of site characterization plan.

The site characterization plan shall contain...(2).(v) Plans to apply quality assurance to data collection, recording and record retention.

60.18 Review of site characterization activities.

(g) During the conduct of site characterization activities, DOE shall report not less than every six months to the Commission on the nature and extent of such activities...DOE shall also report its progress in developing the design of a geological repository operations area.

60.21 Content of application.

(c) The Safety Analysis Report shall include:

(3) A description and analysis of the design and performance requirements for structures systems, and components of the geologic repository which are important to safety. This analysis shall consider (i) The margins of safety under normal conditions and under conditions that may result from anticipated operational occurrences, including those of natural origin; and (ii) the adequacy of structures, systems, and components provided for the prevention of accidents and mitigation of the consequences of accidents, including those caused by natural phenomena.

(6) An identification and justification for the selection of those variables, conditions, or other items which are determined to be probable subjects of license specifications. Special attention shall be given to those items that may significantly influence the final design.

(7) A description of the program for control and monitoring of radioactive effluents and occupational radiation exposures to maintain such effluents and exposures in accordance with the requirements of Part 20 of this chapter.

(8) A description of the controls that the applicant will apply to restrict access and to regulate land use at the site and adjacent areas, including a conceptual design of monuments which would be used to identify the postclosure controlled area after permanent closure.

(9) Plans for coping with radiological emergencies at any time prior to permanent closure and decontamination or dismantlement of surface facilities.

(10) A description of the program to be used to maintain the records described in Secs. 60.71 and 60.72.

(12) A description of plans for retrieval and alternate storage of the radioactive wastes should the geologic repository prove to be unsuitable for disposal of radioactive wastes.

(15) The following information concerning activities at the geologic repository operations area:

(i) The organizational structure of the DOE as it pertains to construction and operation of the geologic repository operations area including a description of any delegations of authority and assignments of responsibilities, whether in the form of regulations, administrative directives, contract provisions, or otherwise.

(ii) Identification of key positions which are assigned responsibility for safety at and operation of the geologic repository operations area.

(iii) Personnel qualifications and training requirements.

(iv) Plans for startup activities and startup testing.

(v) Plans for conduct of normal activities, including maintenance, surveillance, and periodic testing of systems, structures, and components of the geologic repository operation area.

(vi) Plans for permanent closure and plans for the decontamination or dismantlement of surface facilities

(vii) Plans for any uses of the geologic repository operations area for purposes other than disposal of radioactive wastes, with an analysis of the effects, if any, that such uses may have upon the operation of the structures, systems, and components important to safety and the engineered and natural barriers important to waste isolation.

Construction Authorization

60.31 Construction authorization.

Upon review and consideration of an application and environmental impact statement submitted under this part, the Commission may authorize construction if it determines:

(a) Safety. That there is reasonable assurance that the types and amounts of radioactive materials described in the application can be received, possessed, and disposed of in a geologic repository operations area of the design proposed without unreasonable risk to the health and safety of the public. In arriving at this determination, the Commission shall consider whether:

(1) The DOE has described the proposed geologic repository including but not limited to:

(iv) construction procedures which may affect the capability of the geologic repository to serve its intended function; and

(4) The DOE personnel training program complies with the criteria contained in Subpart H of this part.

(5) The DOE emergency plan complies with the criteria contained in Subpart I of this part.

(6) The DOE proposed operating procedures to protect health and to minimize danger to life or property are adequate.

License Issuance and Amendment

60.42 Conditions of license.

A license issued pursuant to this part shall include such conditions, including license specifications, as the Commission finds to be necessary to protect the health and safety of the public, the common defense and security, and environmental values.

60.43 License specification.

A license issued under this part shall include license conditions derived from the analyses and evaluations included in the application, including amendments made before a license is issued, together with such additional conditions as the Commission finds appropriate.

License conditions shall include items in the following categories:

Restrictions as to the physical and chemical form and radioisotopic content of radioactive waste.

Restrictions as to size, shape, and materials and methods of construction of radioactive waste packaging.

Restrictions as to the amount of waste permitted per unit volume of storage space considering the physical characteristics of both the waste and the host rock.

Requirements relating to test, calibration, or inspection to assure that the foregoing restrictions are observed.

Controls to be applied to restricted access and to avoid disturbance to the postclosure controlled area and to areas outside the controlled area where conditions may affect isolation within the controlled area.

(6) Administrative controls, which are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure that activities at the facility are conducted in a safe manner and in conformity with the other license specifications.

60.44 Changes, tests, and experiments.

(a)(1) Following authorization to receive and possess source, special nuclear, or byproduct material at a geologic repository operations area, the DOE may:

(i) make changes in the geologic repository operations area as described in the application,

(ii) make changes in the procedures as described in the application, and

(iii) conduct tests or experiments not described in the application, without prior Commission approval, provided the change, test, or experiment involves neither a change in the license conditions incorporated in the license nor an unreviewed safety question.

(2) A proposed change, test, or experiment shall be deemed to involve an unreviewed safety question if:

(i) the likelihood of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the application is increased,

(ii) the possibility of an accident or malfunction of a different type than any previously evaluated in the application is created, or

(iii) the margin of safety as defined in the basis for any license condition is reduced.

(b) The DOE shall maintain records of changes in the geologic repository operations area and of changes in procedures made pursuant to this section, to the extent that such changes constitute changes in the geologic repository operations area or procedures as described in the application. Records of tests and experiments carried out pursuant to paragraph (a) of this section shall also be maintained. These records shall include a written safety evaluation which provides the basis for the determination that the change, test, or experiment does not involve an unreviewed safety question. The DOE shall prepare annually, or at such shorter intervals as may be specified in the license, a report containing a brief description of such changes, tests, and experiments, including a summary of the safety evaluation of each. The DOE shall furnish the report to the appropriate NRC Regional Office shown in Appendix D of part 20 of this chapter with a copy to the Director, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. Any report submitted pursuant to this paragraph shall be made a part of the public record of the licensing proceedings.

60.46 Particular activities requiring license amendment.

(a) Unless expressly authorized in the license, an amendment of the license shall be required with respect to any of the following activities:

(1) Any action which would make emplaced high-level radioactive waste irretrievable or which would substantially increase the difficulty of retrieving such emplaced waste.

(2) Dismantling of structures.

(3) Removal or reduction of controls applied to restrict access to or avoid disturbance of the controlled area and to areas outside the controlled area where conditions may affect isolation within the controlled area.

(4) Destruction or disposal of records required to be maintained under the provisions of this part.

(5) Any substantial change to the design or operating procedures from that specified in the license.

(6) Permanent closure.

(7) Any other activity involving an unreviewed safety question.

(b) An application for such an amendment shall be filed, and shall be reviewed, in accordance with the provisions of Sec. 60.45.

Permanent Closure

60.51 License amendment for permanent closure.

(a) The DOE shall submit an application to amend the license prior to permanent closure. The submission shall consist of an update of the License Application submitted under Sec. 60.21 and 60.22, including:

(2) A detailed description of the measures to be employed - such as land use controls, construction of monuments, and preservation of records - to regulate or prevent activities that could impair the long-term isolation of emplaced waste within the geologic repository and to ensure that relevant information will be preserved for the use of future generations. As a minimum, such measures shall include:

(i) Identification of the controlled area and geologic repository operations area by monuments that have been designed, fabricated, and emplaced to be as permanent as is practicable; and

(ii) Placement of records in the archives and land record systems of local State, and Federal government agencies, and archives elsewhere in the world, that would be likely to be consulted by potential human intruders - such records to identify the location of the geologic repository operations area, including the underground facility, boreholes and shafts, and the boundaries of the controlled area, and the nature and hazard of the waste.

Subpart D - Records, Reports, Tests, and Inspections

60.71 Records and reports.

(a) The DOE shall maintain such records and make such reports in connection with the licensed activity as may be required by the conditions of the license or by rules, regulations, and orders of the Commission as authorized by the Atomic Energy Act and the Energy Reorganization Act.

(b) Records of the receipt, handling, and disposition of radioactive waste at a geologic repository operations area shall

contain sufficient information to provide a complete history of the movement of the waste from the shipper through all phases of storage and disposal. The DOE shall retain these records in a manner that ensures their usability for future generations in accordance with Sec. 60.51(a)(2).

60.72 Construction records.

(a) The DOE shall maintain records of construction of the geologic repository operations area in a manner that ensures their usability for future generations in accordance with Sec. 60.51(a)(2).

(b) The records required under paragraph (a) shall include at least the following:

(1) Surveys of the underground facility excavations, shafts, and boreholes referenced to readily identifiable surface features or monuments;

(2) A description of the materials encountered;

(3) Geologic maps and geologic cross sections;

(4) Locations and amount of seepage;

(5) Details of equipment, methods, progress, and sequence of work;

(6) Construction problems;

(7) Anomalous conditions encountered;

(8) Instrument locations, readings, and analysis;

(9) Location and description of structural support systems;

(10) Location and description of dewatering systems; and

(11) Details, methods of emplacement, and location of seals used.

60.73 Reports of deficiencies.

The DOE shall promptly notify the Commission of each deficiency found in the characteristics of the site, and design and construction of the geologic repository operations area which, were it to remain uncorrected, could:

(a) Be a substantial safety hazard,

(b) represent a significant deviation from the design criteria and design bases stated in the application, or

(c) represent a deviation from the conditions stated in the terms of a construction authorization or the license, including license specifications. The notification shall be in the form of a written report, copies of which shall be sent to the Director and to the appropriate Nuclear Regulatory Commission Regional Office listed in Appendix D of part 20 of this chapter.

Subpart G – Quality Assurance

60.151 Applicability.

The quality assurance program applies to all systems, structures, and components important to safety, to design and characterization of barriers important to waste isolation, and to activities related thereto. These activities include: site characterization, facility and equipment construction, facility operation, performance confirmation, permanent closure, and decontamination and dismantling of surface facilities.

60.152 Implementation.

The DOE shall implement a quality assurance program based on the criteria of Appendix B to 10 CFR 50 as applicable, and appropriately supplemented by additional criteria as required by 60.151.

Subpart H - Training and Certification of Personnel

60.160 General requirements.

Operations of systems and components that have been identified as important to safety in the Safety Analysis Report and in the license shall be performed only by trained and certified personnel or by personnel under the direct visual supervision of an individual with training and certification in such operation. Supervisory personnel who direct operations that are important to safety must also be certified in such operations.

60.161 Training and certification program.

The DOE shall establish a program for training, proficiency testing, certification and requalification of operating and supervisory personnel.

60.162 Physical requirements.

The physical condition and the general health of personnel certified for operations that are important to safety shall not be such as might cause operational errors that could endanger the public health and safety. Any condition which might cause impaired judgment or motor coordination must be considered in the selection of personnel for activities that are important to safety. These conditions need not categorically disqualify a person, so long as appropriate provisions are made to accommodate such conditions.

Subpart I - Emergency Planning Criteria [RESERVED]
10 CFR Part 70, Domestic Licensing of Special Nuclear Material
70.22 Contents of applications.

(i)(1) Each application to possess enriched uranium or plutonium for which a criticality accident alarm system is required, uranium hexafluoride in excess of 50 kilograms in a single container or 1000 kilograms total, or in excess of 2 curies of plutonium in unsealed form or on foils or plated sources, must contain either:

An evaluation showing that the maximum dose to a member of the public offsite due to a release of radioactive materials would not exceed 1 rem effective dose equivalent or an intake of 2 milligrams of soluble uranium, or

An emergency plan for responding to the radiological hazards of an accidental release of special nuclear material and to any associated chemical hazards directly incident thereto.

Emergency plans submitted under paragraph (i)(1)(ii) of this section must include the following information:...

10 CFR Part 71, Packaging and Transportation of Radioactive Material
Subpart H, Quality Assurance

10 CFR Part 72, Licensing Requirements For The Independent Storage Of Spent Nuclear Fuel And High-Level Radioactive Waste, Physical Protection
72.24 Contents of application: technical information.

Each application for a license under this part must include a Safety Analysis Report describing the proposed ISFSI or MRS for the receipt, handling, packaging, and storage of spent fuel or high-level radioactive waste, including how the ISFSI or MRS will be operated. The minimum information to be included in this report must consist of the following:

A description of the applicant's plans for coping with emergencies, as required by § 72.32.

IV. ACCEPTANCE CRITERIA AND GUIDANCE

General Guidance

This Section contains guidance for preparation of the License Application for construction authorization.

Some guidance is provided in this chapter by reference to NRC regulatory guides and industry codes and standards. Unless otherwise noted in the references to these documents, applicability

of these documents and how to address them has not been fully determined. A preliminary assessment of the applicability of the NRC regulatory guides has been completed and incorporated into this guidance. Other potentially applicable documents (for example, industry codes and standards and NUREGs) that may be useful are identified. However, a formal assessment of the applicability of these additional guidance documents has not been incorporated into this version of the technical guidance document; applicable instructions for the License Application author will be provided in a future revision.

The guidance in this chapter is generally based on or extracted from the following documents:

- NUREG-1323, *License Application Review Plan for a Geologic Repository for Spent Nuclear Fuel and High-Level Radioactive Waste* (NRC 1995a)
- YMP/97-01Q, *Management Plan for the Development of a Project Integrated Safety Assessment* (PISA) (YMP 1997)
- NUREG-1567, *Standard Review Plan for Spent Fuel Dry Storage* (NRC 1996b)
- Regulatory Guide DG-3003, *Format and Content for the License Application for the High-Level Waste Repository* (NRC 1990)

NUREG-1323 (NRC 1995a) is largely incomplete and will become inapplicable to Yucca Mountain upon the expected issuance of a new 10 CFR 63 as discussed in Section III above. Information from NUREG-1323 (NRC 1995a) considered likely to be applicable under the new regulation is included in the technical guidance document, though the NUREG is not referenced as an authoritative source for such information. The *Management Plan for the Development of a Project Integrated Safety Assessment (PISA)* (YMP 1997) contains the Project Integrated Safety Assessment Content Guide that has been developed by the Yucca Mountain Site Characterization Project. The Project Integrated Safety Assessment Content Guide is the starting point for guidance for preparing the License Application. The *Standard Review Plan for Spent Fuel Dry Storage Facilities* (NRC 1996b), although not directly applicable to the repository or the waste package, pertains to containers for dry storage of spent nuclear fuel and therefore contains much useful and transferable information for developing the License Application. This document contains detailed information that is missing from other NRC guidance documents for the repository. As is the case for NUREG-1323 (NRC 1995a), DG-3003 (NRC 1990) will become inapplicable to Yucca Mountain upon the expected issuance of a new 10 CFR 63 as discussed in Section III above. Information from DG-3003 (NRC 1990) considered likely to be applicable under the new regulation is included in the technical guidance document, though DG-3003 is not referenced as an authoritative source for such information.

In addition to the documents identified above, the information in YMP/94-05, *Mined Geologic Disposal System License Application Annotated Outline* (DOE Office of Civilian Radioactive Waste Management, in draft), was used extensively to develop the topics and regulatory interfaces contained herein.

11.0 PURPOSE, SUMMARY, AND CONCLUSIONS

Provide an introductory discussion of the purpose of Chapter 11 of the License Application. Include a summary of the information contained in the chapter and a discussion of the organization of the chapter. Provide a summary of conclusions supporting the intent of this chapter and a description of important uncertainties, if applicable, which affect the conclusions.

11.1 CONDUCT OF OPERATIONS

Describe (in more detail than in Chapter 1) plans for conduct of normal operations activities at the geologic repository operations area as required in 10 CFR 60.21(c)(15)(v), and plans for conduct of normal activities including maintenance, surveillance, and periodic testing of structures, systems, and components of the geologic repository operation areas. Address the activities that are expected to occur relatively frequently (daily, weekly, monthly, annually). This section is not concerned with other activities, such as performance confirmation, which are discussed elsewhere in the License Application. Discuss operations involving all those geologic repository operations area facilities that ensure the safe handling of radioactive materials during normal repository operations activities, such as startup and testing, waste receipt, temporary waste storage, waste transfer, waste emplacement, and routine geologic repository operations area inspection and testing. Address maintenance, surveillance, and periodic testing of structures, systems, and components important to safety or retrievability. Discuss interfaces between geologic repository operations area operators and the geologic repository operations area physical facility that permit the conduct of routine activities.

As required by 10 CFR 60.21(c)(15)(vii), discuss any activities to be conducted at the facility for purposes other than disposal of radioactive waste. Create a subsection labeled "Activities Not Related to Disposal of Radioactive Wastes," and describe the activities. Include the project effects of all such activities on waste isolation and repository performance. Note that activities in support of radioactive waste disposal, even if only indirectly related to such activities, are not within the scope of this regulation.

As required by 10 CFR 19.11 and 10 CFR 19.12, discuss required posting of notices and instructions to workers.

Demonstrate that conduct of operations is in accordance with the following regulatory guides, as appropriate to the topical area under consideration, and the Monitored Geologic Repository (MGR) system characteristics.

NOTE: Guidance identified in this section involves quality assurance references other than the *Quality Assurance Requirements and Description* (QARD) manual (DOE 1998). The QARD is the governing quality assurance document for the Yucca Mountain Site Characterization Project (see Chapter 14), but the guidance identified here provides more specific information than that provided in the QARD to assist the author in preparing this section.

- Regulatory Guide 1.23, *Onsite Meteorological Programs* (NRC 1972a), describes a suitable onsite meteorological program to provide data needed to estimate potential

radiation doses. Section C, Regulatory Position, includes guidance applicable to instrument maintenance and servicing, including schedule, and shall be considered in describing the conduct of operations. In particular, Section C (5) (Instrument Maintenance and Servicing Schedules), shall be considered to be applicable.

- Regulatory Guide 1.12, *Nuclear Power Plant Instrumentation for Earthquakes* (NRC 1997), describes earthquake instrumentation acceptable to NRC, and briefly notes where mitigating procedures should be available in the event of an earthquake.

- Regulatory Guide 1.116, *Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems* (NRC 1976b), documents that the requirements of ANSI N45.2-1975, *Supplementary Quality Assurance Requirements for the Installation, Inspection, and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Power Plants* (ANSI/ANS 1975), are an acceptable approach for satisfying 10 CFR 50, Appendix B. In addition, NRC specifies that the provisions of the referenced standard apply to operation as well as construction. Sections C (1), (2), and (3) shall be considered applicable to the MGR, as appropriate, for mechanical equipment that is important to safety.

- Regulatory Guide 1.118, *Periodic Testing of Electric Power and Protection Systems* (NRC 1995c), states, with exceptions, that IEEE Standard 338-1987, *Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems* (IEEE 1987), is an acceptable approach for conducting periodic testing of electric power and protection systems. The regulatory positions stated in Sections C (1), (2), and (3) shall be considered for testing electrical systems in the MGR that are important to safety.

- Regulatory Guide 1.128, *Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants* (NRC 1978c), identifies IEEE Standard 484-1975, *IEEE Recommended Practice for Installation Design and Installation of Large Lead Storage Batteries for Generating Stations and Substations* (IEEE 1975a), as acceptable practice. (IEEE 1975a has also been designated ANSI N41.24-1976). To the extent that large lead storage batteries are incorporated in the MGR in systems important to safety, the provisions shall be considered applicable.

- Regulatory Guide 1.129, *Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants* (NRC 1978d), identifies IEEE Standard 450-1975, *IEEE Recommended Practice for Maintenance, Testing and Replacement of Large Lead Storage Batteries for Generating Stations and Substations* (IEEE 1975b), as an acceptable basis for demonstrating compliance. If large lead storage batteries are incorporated in the MGR in systems import to safety, regulatory positions C (1) and C (2) of this guide will be considered applicable.

- Regulatory Guide 1.33, *Quality Assurance Program Requirements (Operation)* (NRC 1987b), references ANSI N18.7-1976, *Administrative Controls and Quality Assurance/Nuclear* (ANSI/ANS 1976), as an acceptable approach for satisfying

10 CFR 50, Appendix B, quality assurance requirements. Particular emphasis is placed on the need for procedures. The detailed guidance in this guide is specific to nuclear stream supply system systems, components, and events. However, consideration of selected portions of the guidance is appropriate for the repository. Regulatory positions C (1) and C (5)(a) shall be considered to be applicable. The following sections of Appendix A to this guide may also be applicable to the extent that relevant systems and components important to safety are incorporated in the MGR design: A1; A2k; A3h, m, q, r, s; A4k; A5; A6; A7a, b, e, f; 8; and 9.

- Regulatory Guide 1.39, *Housekeeping Requirements for Water-Cooled Nuclear Power* Plants (NRC 1977a), recognizes ANSI N45.2.3-1973, *Housekeeping During the Construction Phase of Nuclear Power Plants* (ANSI/ANS 1973), as acceptable for satisfying 10 CFR 50, Appendix B, requirements. In addition, NRC has defined the scope of applicability to include operation as well as construction where similar activities may be undertaken. Sections C (1) and C (3) of this guide shall be considered to be applicable to MGR construction activities, including construction that occurs during operations.

- Regulatory Guide 1.68, *Initial Tests (sic) Programs for Water-Cooled Nuclear Power Plants* (NRC 1978b), defines the scope of a graded test program to be conducted before issuance of a full-power operating license. Regulatory positions stated at C1c, d, e, and f; and C2 through C9 shall be considered to be generally applicable to the development of repository preoperational testing programs for systems and components important to safety. In addition, Appendix A1, (m) (1)-(4), of this guide shall be considered applicable.

- Regulatory Guide 1.147, *Inservice Inspection Code Case Acceptability ASME Section XI Division 1* (NRC 1994), documents that Section XI of the ASME *Boiler and Pressure Vessel Code* is acceptable for demonstrating compliance, with some exceptions. To the extent that inservice inspections (using ultrasonic, eddy current, magnetic particle, and other techniques discussed in Section XI) of MGR components important to safety are conducted, Sections C (1) and (2) of this guide shall be considered applicable unless equally effective alternatives, acceptable to NRC, are identified.

- Regulatory Guide 1.160, *Monitoring the Effectiveness of Maintenance at Nuclear Power Plants* (NRC 1995b), is specific to licensees subject to 10 CFR 50.65 (requirements for monitoring the effectiveness of maintenance at nuclear power plants). The guidance contains information that is relevant to the conduct of operations at the repository, and Section C shall be considered to be applicable to the monitoring of safety-related engineered repository systems or components, where practicable.

- Regulatory Guide 3.48, *Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage)* (NRC 1986c), Sections 9.2 (Preoperational Testing), 9.3

(Training Programs), and 9.4 (Procedures), shall be considered applicable to Section 11.1.

- Regulatory Guide 1.70, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants LWR Edition* (NRC 1978f), Chapters 13 and 14, contain relevant guidance. Section 3.9.6 (Inservice Testing of Pumps and Valves); Section 6.6 (Inservice Inspection Program of Class 2 and 3 Components); and Chapter 14 (Initial Test Programs) shall be considered to be applicable. Applicability of guidance in this guide should be considered from the perspective of whether or not it duplicates or inappropriately expands on the guidance in NUREG-1323 (NRC 1995a), DG-3003 (NRC 1990), or Regulatory Guide 3.48 (NRC 1986c).

- Regulatory Guide 3.53, *Applicability of Existing Regulatory Guides to the Design and Operation of an Independent Spent Fuel Storage Installation* (NRC 1982), lists those regulatory guides NRC believes to be applicable to an independent spent fuel storage installation even though they were originally developed for other facilities. The following regulatory guides listed in this guide as being applicable to an independent spent fuel storage installation shall also be considered to be applicable to the repository:

  - Regulatory Guide 1.28, *Quality Assurance Program Requirements (Design and Construction)* (NRC 1985)

  - Regulatory Guide 1.108, *Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants* (NRC 1977b)

  - Regulatory Guide 1.22, *Periodic Testing of Protection System Actuation Functions* (regulatory positions 1, 2, and 4) (NRC 1972c)

  - Regulatory Guide 1.143, *Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants* (NRC 1979a). (Applies to wastes generated during operations only.)

The general content of Regulatory Guide 3.53 (NRC 1982) provides valuable insight into NRC thinking relative to the transferability of regulatory guidance to facilities other than those for which the guidance was originally prepared. As such, consideration should be given to the recommendations provided in this guide that extend beyond the documents specifically cited above.

11.1.1 Repository Construction

Describe the repository construction process, including the interfaces between construction and performance confirmation activities and construction and waste emplacement activities when both are proceeding at the same time. The level of detail should be great enough to enable NRC to understand and agree to the validity of the concepts, but not so great as to require License Application updates to address minor changes in the concepts. To the extent this information is

provided in Chapter 4 of the License Application, it may be referenced (and not repeated) here. However, emphasis in this section is on an integrated discussion of the construction process rather than focusing on specific repository configuration items.

11.1.2 Startup Activities and Testing

As required by 10 CFR 60.21(c)(15)(iv), describe the startup activities and testing planned for beginning waste receipt and emplacement in conjunction with receipt of the license to receive and possess such waste. Describe any testing involving "real" waste. Description should include requirements to be verified, safety risks to be addressed, and a detailed explanation of any conditional licensing agreements needed to conduct these tests. Refer to the License Application test and evaluation plan for a more complete description of startup testing. The level of detail should be great enough to enable NRC to understand and agree to the validity of the concepts, but not so great as to require License Application updates to address minor changes in the concepts.

11.1.3 Waste Handling Activities

Describe in general terms the activities that occur during waste handling and reference appropriate sections of Chapter 4 for more detailed information. The level of detail should be great enough to enable NRC to understand and agree to the validity of the concepts, but not so great as to require License Application updates to address minor changes in the concepts. Address the following topics:

- Waste receipt and inspection (including interface with outside organizations, such as regional service contractors, if used)
- Temporary waste storage
- Waste handling activities in the surface facilities
- Waste handling in shafts and ramps
- Waste handling (including emplacement) in the underground facility

11.1.4 Surveillance and Periodic Testing

Briefly describe the equipment surveillance/periodic testing program for equipment important to safety and to waste isolation. Include a description of the purpose of the program and the manner in which the surveillance requirements were or will be developed. Typically, the surveillance requirements for equipment important to safety are contained in the technical specifications, a separate document. Refer the reader to the technical specifications, the License Application test and evaluation plan, and other project documents, as appropriate, for a more complete description of the surveillance/periodic testing program.

Section IX of the ASME *Boiler and Pressure Vessel Code* may provide applicable guidance and requirements for the surveillance program for equipment important to safety. Identify the latest

ASME Section XI version accepted by NRC at the time of writing the License Application to establish the proper reference. This information is found in 10 CFR 50.55a(b)(2).

11.1.5 Maintenance

Provide a description of the facility maintenance plan. Introduce the maintenance section by including a brief summary of each major subsection. State that this section will show that facility maintenance is conducted in an effective manner and will ensure the reliable operation of repository systems, structures, and components.

11.1.5.1 Maintenance Program Policy, Goals, and Objectives

State the overall facility policy or policies for the conduct of maintenance. Show how these policies establish the groundwork for effective implementation of maintenance to meet program objectives.

State the objectives of the facility maintenance program. Discuss equipment degradation, failure, reliability and the ability of systems, structures, and components to perform design functions.

Generally describe how quantitative maintenance goals are established and used to achieve program objectives. Mention the system for maintaining equipment history and reliability information. (Maintenance record keeping is discussed in Section 11.1.5.2.) The discussion should be general and cover equipment monitoring, reliability, and history. Specific maintenance goals are not required.

11.1.5.2 Conduct of Maintenance

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—Refer to Section 11.8 for training program details.

Maintenance Support Organizations—Describe the planned interface between the maintenance organization and other support organizations, including operations, health physics, and engineering. Show how these organizations support an effective maintenance program in:

- Engineering–regulatory and design requirements, procurement specifications, test requirements, and root cause evaluations
- Operations–operability requirements and technical specifications
- Health Physics–radiation control and mock-up training

Maintenance Procedures—Identify and describe the operational procedures for maintenance of systems, structures, and components 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 systems, structures, and components important to safety, retrievability, and isolation for all surface facilities, shafts, and ramps, and the underground facility. Refer to Section 11.3 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 important systems, structures, and components.

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.

11.1.5.3 Maintenance Program Monitoring, Assessment, and Feedback

Describe the methods planned for assessing the safety and effectiveness of the repository maintenance program. Include a discussion of audits, inspections, and comparison of plant performance against the goals established in Section 11.1.5.1. 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 1995b) (as determined applicable), and provide support for the statement. The maintenance rule requires that the holder of an operating license monitor the performance or condition of systems, structures, and components against licensee-established goals, in a manner sufficient to provide reasonable assurance that they are capable of fulfilling their intended functions. Regulatory Guide 1.160 (NRC 1995b) provides guidance in satisfying the requirements of 10 CFR 50.65.

11.1.6 Waste Retrieval, Removal, and Alternate Storage

Describe the plan for retrieval, removal, and alternative storage of radioactive wastes should the geologic repository prove to be unsuitable for disposal of radioactive wastes. This description is required in accordance with 10 CFR 60.21(c)(12).

11.2 ORGANIZATIONAL STRUCTURE, MANAGEMENT, STAFFING, AND ADMINISTRATIVE CONTROLS

The purpose of this section is to demonstrate that DOE has in place adequate administrative procedures and controls, organization, management, and staff to comply with the requirements of 10 CFR 60 and other regulations referenced in Part 60, and to demonstrate the ability 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 operating organization. This organizational description satisfies the requirements of 10 CFR 60.21(c)(15)(i), (ii), and (iii).

Introduce this section of the License Application by providing a short summary of each major subsection. Specify the phase(s) of facility operation applicable to the organization discussed.

Refer to the *Safety Analysis Report for the Idaho National Engineering Laboratory TMI-2 Independent Spent Fuel Storage Installation* (DOE 1996) and the *Centralized Interim Storage Facility Topical Safety Analysis Report* (DOE 1997) for examples of an organizational description involving both DOE and an M&O.

Demonstrate that organizational structure, management, staffing, and administrative controls are consistent with the regulatory guidance that follows, as appropriate to the topical area under consideration.

Additional evaluation will be performed later to ensure that the guidance presented in the following references is internally consistent, and, more importantly, does not contradict the content of DOE standards referenced later in the text. If there are inconsistencies between DOE standards/orders and NRC guidance, the latter should take precedence.

- The following sections of Regulatory Guide 1.70, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants LWR Edition* (NRC 1978f), shall be considered applicable:
  - 17.1.1, Organization (Quality Assurance)
  - 13.1, Organizational Structure of Applicant
  - 13.2, Training
  - 13.4, Review and Audit
  - 13.5, Plant Procedures
- Section 9.1 and 9.3 of Regulatory Guide 3.48, *Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage)* (NRC 1986c), shall be considered applicable.

11.2.1 U.S. Department of Energy

11.2.1.1 Organization

Describe the organizational structure of DOE offices having jurisdiction over planning, construction, and preoperational aspects of the project, including physical security. Provide an organizational diagram showing the structure and position titles. Show clear lines of authority, oversight, responsibility, and the ability of DOE to exercise management and administrative control of activities at the repository. Include both the offsite and onsite DOE organizations. If the proposed organization is different for various phases of the repository, describe each applicable organization.

11.2.1.2 Personnel Functions, Responsibilities, and Authorities

List the key onsite DOE management personnel and functions, including authorities and responsibilities. Provide a description of delegation of authority (including specific succession to responsibility for overall operation of the repository in the event of absences, incapacitation, or other emergencies) and assignment of responsibility, whether by regulations, administrative directives, contract provisions, or otherwise. At a minimum, include the DOE site manager, quality assurance manager, safety and health officer, and technical manager or equivalents.

11.2.1.3 Personnel Qualification Requirements

Describe the proposed minimum qualification requirements (experience, training, and education) for specific functions or positions described in Section 11.2.1.2. The minimum qualification requirements should be stated for major operating, technical, health physics, and maintenance supervisory personnel, including major contractors. The qualifications of the individuals assigned to the managerial and technical positions described should be presented in resume form. The resumes should identify individuals by position title and, at a minimum, should describe the formal education, training, and pertinent experience of the individuals.

11.2.1.4 Interrelationships with Contractors

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 between the organizations to ensure safe operation of the repository (NRC 1986c, Section 9.1.1.3).

11.2.1.5 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. Also include agreements with security or police agencies, medical facilities, fire departments, emergency response teams, and any other governmental agencies (NRC 1986c, Section 9.1.4).

11.2.2 Management and Operating Contractor

11.2.2.1 Management and Operating Contractor In-House Organization

Describe the organizational structure of the major contractors with responsibility for management and operational control. Include the major contractors that are working on design, preparation, or are projected to have roles in the construction or operation of the repository. Provide an organizational diagram and show clear lines of authority, oversight, and responsibility. If the proposed organization is different for various phases of the repository, describe each applicable organization.

Personnel Functions, Responsibilities, and Authorities—List the key onsite management personnel and functions, including authorities and responsibilities. Describe the procedure documenting changes in both identification and qualifications of key personnel. 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.

11.2.2.2 Technical Staff

Describe the M&O technical staff supporting facility siting, engineering, construction, and operations. Provide a description of the technical staff duties, responsibilities and authorities. 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 1986c, Section 9.1.1.4). Guidelines for technical support organizations are provided in ANSI/ANS-3.2-1994 (ANSI/ANS 1994). Establish that the experience and qualification of technical staff personnel meet the requirements of ANSI/ANS-3.1-1993 (ANSI/ANS 1993).

11.2.2.3 Management and Operating Contractor Operating Organization

Onsite Organization—Describe the organization responsible for safe operation of the repository. Include key supervisory personnel and those operating positions requiring specific qualifications and/or certifications.

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.

11.2.3 Administrative Controls

Provide a description of the administrative controls relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to ensure that activities at the facility are conducted in a safe manner and in conformity with the other license specifications. This section satisfies the requirements of 10 CFR 60.43(b)(6).

11.2.3.1 Unreviewed Safety Questions

Discuss the program in place to review and document facility modifications. State that the program satisfies the requirements of 10 CFR 60.44. (For additional information to support the program description, see 10 CFR 50.59 or 10 CFR 72.48.)

11.2.3.2 Configuration Management

Provide a brief description of the repository configuration management program. Specifically address configuration control as it relates to software, model, and design controls during the various phases of the project. State how the configuration management program results in safe operation of the repository. State that the facility configuration management satisfies the guidance provided in DOE-STD-1073-93-Part 1 and Part 2 (DOE 1993).

11.2.3.3 Audits, Appraisals, and Reviews

Describe the audits, appraisals, and reviews that are performed on a routine basis to ensure compliance with regulatory and safety requirements at the facility. State that the reviews and audits described above satisfy the guidance provided in ANSI/ANS-3.2-1994 (ANSI/ANS 1994).

Identify and discuss the key committees to provide oversight, evaluations, and recommendations to project management on protection of the environment, public, and personnel safety and health. Include both a facility or site safety review committee and a corporate or management safety review committee. Identify the makeup of each committee's members and alternates, selection method, period of committee service, minimum education, experience, and qualifications. Discuss the schedule for regular meetings, committee responsibilities and authorities, protocol for evaluation of incidents reportable to NRC, established evaluations, reports and follow-up requirements, and report retention requirements. State that the facility and corporate safety review committees satisfy the guidance in ANSI/ANS-3.2-1994 (ANSI/ANS 1994).

11.2.3.4 Employee Concerns Program

Briefly describe the employee concerns program in place at the time the License Application is written and planned for construction and operation. State that the program is documented by appropriate procedure and that employees receive formal training on the program. State that an NRC Form 3 will be posted as required by 10 CFR 60.9.

11.3 PROCEDURES

Procedural requirements for a nuclear facility ultimately stem from the mandate of 10 CFR 50, Appendix B, to establish a quality assurance program that provides control over activities

affecting the quality of the identified structures, systems, and components to an extent consistent with their importance to safety. One programmatic requirement is to establish control of the issuance of documents, such as instructions, procedures, and drawings, including changes thereto, which prescribe all activities affecting quality.

This section will describe the procedural administrative control process(es) for developing, reviewing, approving, controlling, and changing all classifications of procedures. It will describe the plan(s) for developing the procedures that will be used for the programmatic implementation of quality assurance requirements that will be used 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. Introduce this section by summarizing each major subsection (11.3.1, 11.3.2, and 11.3.3).

11.3.1 Procedural Development, Approval, and Change Control

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 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.

NOTE: Guidance identified in this section involves quality assurance references other than the QARD manual (DOE 1998). The QARD is the governing quality assurance document for the Yucca Mountain Site Characterization Project (see Chapter 14), but the guidance identified here provides more specific information than that provided in the QARD to assist the author in preparing this section.

Additional quality assurance regulations concerning development of a procedural process for administrative and implementing procedures is provided in, but not limited to, the following documents:

- 10 CFR 50, Appendix B; Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants
- 10 CFR 60, Subpart G; Quality Assurance
- 10 CFR 71, Subpart H; Quality Assurance
- 10 CFR 72, Subpart G; Quality Assurance
- 10 CFR 73, Physical Protection of Plants and Materials

The following regulatory guides shall be used in developing the procedural requirements for the repository:

- Regulatory Guide 1.33, *Quality Assurance Program Requirements (Operation)* (NRC 1978e), describes a method acceptable to NRC staff for complying with the NRC regulations with regard to overall quality assurance program requirements for the operation phase of nuclear power plants. This guide and, by inclusion, ANSI/ANS 1976 and ANSI/ANS 1994, provides guidance to ensure minimum procedural coverage for plant operating activities, including maintenance activities, that should be covered by written procedures. Many of these requirements are generally applicable to spent fuel and high-level radioactive waste handing activities.

- Regulatory Guide 3.48, *Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage)* (NRC 1986c), describes a standard format acceptable to NRC staff for the submittal of a Safety Analysis Report during License Application for these types of installations. Sections 9.4 and 9.4.1 require that the applicant make a commitment to conduct operations that are important to safety in accordance with written procedures and a programmatic procedure review, change, and approval process.

- Regulatory Guide 3.50, *Standard Format and Content for a License Application to Store Spent Fuel and High-Level Radioactive Waste* (NRC 1989), Chapter 5, Operating Procedures, states that NRC staff must find that the applicant's proposed operating procedures to protect health and to minimize danger to life or property are adequate.

NOTE: Although the following draft regulatory guides are included here for consideration, they will be evaluated for applicability when they are formally issued as regulatory guides.

- DG-1010, *Quality Assurance Program Requirements (Design, Construction, Modification and Decommissioning)*, proposed Rev. 4 to Regulatory Guide 1.28 (NRC 1985), extends the scope to apply to modifications and decommissioning.

- DG-RS 902-4, *Quality Assurance Program Requirements (Operation),* second proposed Rev. 3 to Regulatory Guide 1.33 (NRC 1978e), contains revised guidance affecting quality assurance programs initiated by revision to ANSI N18.7-1976/ANS-3.2-1994 (ANSI/ANS 1994) and changes in philosophy ("lessons learned") due to development by NRC and various other organizations.

- DG-1064, *An Approach for Plant-Specific, Risk-Informed Decision Making: Graded Quality Assurance*, provides potential changes to quality assurance requirements by incorporating the concept of graded quality assurance. This concept is intended to provide a safety (and cost) benefit by allowing NRC and licensees to preferentially allocate resources based on the safety significance of the item.

11.3.2 Administrative Procedures

This section will describe the plan(s) for identifying, developing, and implementing the upper tier administrative process controls and program descriptions and procedures to be used to guide the conduct of quality activities affecting the geologic repository. Details of procedures are not required.

Since the design and construction phase and the operational phase of a nuclear facility requires differing quality assurance emphases, NRC has issued two regulatory guides for nuclear power plants that differentiate these phases.

The following regulatory guides shall be used in developing the administrative procedure requirements for the repository:

NOTE: Guidance identified in this section involves quality assurance references other than the QARD manual (DOE 1998). The QARD is the governing quality assurance document for the Yucca Mountain Site Characterization Project (see Chapter 14), but the guidance identified here provides more specific information than that provided in the QARD to assist the author in preparing this section.

- Regulatory Guide 1.28, *Quality Assurance Program Requirements (Design and Construction)* (NRC 1985), specifies acceptable implementation methods for quality-related activities affecting structures, systems, and components during the design and construction phase of a nuclear facility. These activities include designing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying such structures, systems, and components. This guide addresses and endorses those portions of ASME NQA-1-1983 (ASME 1989) applicable to the design and construction phase of nuclear power plants.

- Regulatory Guide 1.33, *Quality Assurance Program Requirements (Operation)* (NRC 1978e). This guide specifies acceptable implementation for quality-related activities affecting structures, systems, and components during the operational phase of a nuclear facility. These activities include operating, surveilling, testing, maintaining, repairing and modifying in-service structures, systems, and components that may be providing a safety function. This guide endorses ANSI N18.7-1976/ANS 3.2-1994 (ANSI/ANS 1994) and Appendix A lists typical safety related activities that should be covered by written procedures (but does not provide a complete listing). The procedures listed in Appendix A may be combined, separated, added to, or deleted to conform to the applicant's procedures plan and facility configuration.

- Regulatory Guide 3.48, *Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage)* (NRC 1986c), recommends that an applicant should make a commitment to conduct operations that are important to safety in accordance with detailed written procedures. In terms of administrative procedures, it also includes a

requirement that the review, change, and approval practices (processes) be proceduralized.

It is recognized that the procedural requirements for a nuclear power plant differ substantially from those that will be necessary for a repository. However, in lieu of repository-specific requirements, application of the above regulatory guidance will indicate where procedures are necessary to implement quality-related controls.

Give a general description and listing of the administrative and programmatic procedures that will be used to provide control of activities considered to be quality affecting and/or important to safety. Typical upper-tier administrative/program procedure categories may include, but are not limited to, the following:

1. Personnel-Related Topics
   - Personnel policies
   - Timekeeping
   - Overtime/call-in policy
   - Organization
   - Roles and responsibilities
2. Configuration Management
   - Procedure review and approval
   - Drawing control
   - Configuration change mechanisms
   - Record retention
   - Document storage requirements
3. Engineering and Technical Programs
   - Technical design process
   - Design modification control
   - Temporary modification
   - Maintenance of current licensing basis
   - Maintenance of facility design basis
4. Maintenance Program
   - Maintenance policies
   - Housekeeping
   - Work control program
   - Maintenance and test equipment
   - Maintenance rule applications
5. Conduct of Operations
   - Shift staffing
   - Shift turnover
   - Equipment control

- Operating procedures
- Emergency procedures

6. Radiation and Environmental Protection

7. Quality Assurance

8. Corrective Action Program

9. Training

10. Security, etc.

Provide a schedule for the prioritized preparation of appropriate written administrative procedures that ensure regulatory compliance with quality assurance requirements at this facility.

11.3.3 Design and Construction/Operating Procedures

This section will describe the plan(s) for identifying and developing the detailed implementing procedures to be used during the conduct of normal business and operation of the facility from initial design and construction through the preclosure period. These efforts should focus on activities that can reasonably be expected to occur on a routine basis. Special and/or one-of-a-kind procedures should be developed as the need arises. Details of the procedures are not required.

During the design and construction phase, and in accordance with the requirements of Regulatory Guide 1.28 (NRC 1985), implementing procedures should focus on the quality-affecting activities involved with design, purchasing, fabrication, handling, shipping, storage, cleaning, erecting, installing, inspecting, testing, maintaining, repairing, and modifying structures, systems, and components which will perform or contribute to a safety function. Preoperational testing and equipment run-in is considered part of the construction phase. Additional non-quality procedures may be useful to promote uniform work practices and formalize the conduct of normal business.

In addition to the regulatory guidance discussed in Sections 11.3.1 and 11.3.2 for administrative process controls, the following regulatory guides will apply in part while preparing design and construction phase implementing procedures for the repository:

NOTE: Guidance identified in this section involves quality assurance references other than the QARD manual (DOE 1998). The QARD is the governing quality assurance document for the Yucca Mountain Site Characterization Project (see Chapter 14), but the guidance identified here provides more specific information than that provided in the QARD to assist the author in preparing this section.

- Regulatory Guide 3.53, *Applicability of Existing Regulatory Guides to the Design and Operation of an Independent Spent Fuel Storage Installation* (NRC 1982), provides a listing of existing regulatory guides with potential application during the design and

operating phases. Many of the items listed require procedural control under a quality assurance program.

- Regulatory Guide 1.58, *Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel* (NRC 1980b), describes an acceptable compliance method for the qualification of inspection, examination, and testing personnel for quality affecting construction projects.

- Regulatory Guide 1.30, *Quality Assurance Requirements for the Installation, Inspection and Testing of Instrumentation and Electric Equipment* (NRC 1972b), specifies that the requirements and qualifications of testing personnel apply during the operational phase as well as the construction phase.

- Regulatory Guide 1.37, *Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants* (NRC 1973), specifies quality assurance requirements for onsite cleaning of materials and components, cleanness controls and preoperational cleaning, and lay up of nuclear power plant fluid systems.

- Regulatory Guide 1.38, *Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants* (NRC 1977c), covers material handling requirements during construction of nuclear facilities.

- Regulatory Guide 1.94, *Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants* (NRC 1976c), specifies concrete and steel inspection requirements during construction.

- Regulatory Guide 1.116, *Quality Assurance Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems* (NRC 1976b), specifies that these requirements are applicable during the operational phase as well as construction.

- Regulatory Guide 1.39, *Housekeeping Requirements for Water Cooled Nuclear Power Plants* (NRC 1977a), describes the housekeeping requirements for control of work activities, conditions, and environment at nuclear plants both during construction and operation.

Appendix A to Regulatory Guide 1.33, Typical Procedures to Pressurized Water Reactors and Boiling Water Reactors (NRC 1978e), while not a complete list of required procedures, shall be used as guidance to ensure minimum procedural coverage for repository operating activities, including related test and maintenance activities. During the operational phase of the repository, and in accordance with this guide, implementing procedures should focus on the quality-affecting activities associated with normal operation and the postulated upset/emergency conditions. Startup testing is considered to be within the operational phase. Normal operations would encompass waste receipt, temporary waste storage, waste transfer and transportation, waste emplacement, and potential waste retrieval. Included would be normal maintenance

activities, routine inspection and testing, performance confirmation, and emergency response. Procedures will be required for activities impacting the safety function of quality class structures, systems, and components, and may be desirable for some non-quality activities in order to standardize work practices.

In addition to the regulatory guidance discussed in Sections 11.3.1 and 11.3.2 for administrative process controls, the following regulatory guides will apply in part while preparing operational phase implementing procedures for the repository:

NOTE: Guidance identified in this section involves quality assurance references other than the QARD manual (DOE 1998). The QARD is the governing quality assurance document for the Yucca Mountain Site Characterization Project (see Chapter 14), but the guidance identified here provides more specific information than that provided in the QARD to assist the author in preparing this section.

- Regulatory Guide 1.13, *Spent Fuel Storage Facility Design Basis* (NRC 1975b), specifies design requirements for protection of radioactive materials and safety components. Some of these requirements will translate into procedural/programmatic monitoring and/or protective practices during operation.

- Regulatory Guide 3.57, *Administrative Practices for Nuclear Criticality Safety of Fuels and Materials Facilities* (NRC 1986a), describes administrative practices for the prevention of criticality accidents at facilities other than nuclear power plants.

- Regulatory Guide 3.58, *Criticality Safety for Handling, Storing and Transporting LWR Fuels at Fuels and Materials Facilities* (NRC 1986b), describes procedures acceptable to NRC staff for preventing criticality accidents in operations involving handling, storing, and transporting light water reactor fuels at fuels and materials facilities.

- Regulatory Guide 1.116, *Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems* (NRC 1976b), specifies that the requirements are applicable during the operational phase as well as construction.

- Regulatory Guide 3.53, *Applicability of Existing Regulatory Guides to the Design and Operation of an Independent Spent Fuel Storage Installation* (NRC 1982), provides a listing of existing regulatory guides with potential application during the design and operating phases. Many of the items listed require procedural control under a quality assurance program.

- Regulatory Guide 1.68, *Initial Tests (sic) Programs for Water-Cooled Nuclear Power Plants* (NRC 1978b), specifies the test programs for safety-related structures, systems, and components to ensure performance of safety characteristics.

- Regulatory Guide 1.39, *Housekeeping Requirements for Water-Cooled Nuclear Power Plants* (NRC 1977a), describes the housekeeping requirements for control of

work activities, conditions, and environment at nuclear plants both during construction and operation.

- Regulatory Guide 1.30, *Quality Assurance Requirements for the Installation, Inspection and Testing of Instrumentation and Electric Equipment* (NRC 1972b), specifies that the requirements and qualifications of testing personnel apply during the operational phase as well as the construction phase.

Give a general description and listing of the operating procedures that will be used to provide control of site activities important to the safe operation of the repository. Provide a schedule for the prioritized preparation of appropriate written implementing procedures that ensure regulatory compliance with quality assurance requirements for the operational phase of this facility.

11.4 RECORDS AND REPORTS

This section describes the program established to maintain records and reports of geologic repository construction and operations. Introduce the section by providing a brief summary, including the scope of the system established for maintaining facility records. State that repository records are maintained in accordance with requirements of 10 CFR 60.4 and 10 CFR 60, Subpart D. State that the preservation of records following permanent repository closure is described in Section 11.6.

11.4.1 Records Retention

Regulatory requirements and guidance concerned with initiation and retention of records include, but are not limited to, the following documents:

- 10 CFR 60, Disposal of High-Level Radioactive Wastes in Geologic Repositories, which reads in part:
  - Section 60.17, Contents of Site Characterization Plan: "The site characterization plan shall contain ...(2).(v) Plans to apply quality assurance to data collection, recording and record retention."
  - Section 60.21, Content of Application: "An application shall consist of general information and a Safety Analysis Report...(F)(4) A description of the quality assurance program..."
  - Section 60.31, Construction Authorization: "...the Commission may authorize construction, if...(a)(3) The DOE's quality assurance program complies with the requirements of Subpart G..."
  - Section 60.43, License Conditions: "(b) License Conditions shall include...(b)(6) Administrative controls, which are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure that activities...are conducted in a safe manner..."

- Section 60.44, Changes, Tests and Experiments: "...(b) The DOE shall maintain records of changes in the geologic repository operations area..."

- Section 60.71, Records and Reports: "(a) The DOE shall maintain such records and make such reports in connection with the licensed activity as may be required by the conditions of the license, or by rules, regulations and orders of the Commission..."

- Section 60.72, Construction Records: "(a) The DOE shall maintain records of construction of the geologic repository operations area in a manner that ensures their usability for future generations..."

- Subpart G, Quality Assurance, Section 60.152: "The DOE shall implement a quality assurance program based on the criteria of Appendix B to 10 CFR 50..."

- 10 CFR 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants, which reads in part:

  - Subsection XVII, Quality Assurance Records: "Sufficient records shall be maintained to furnish evidence of activities affecting quality. The records shall include at least the following: operating logs and the results of reviews, inspections, tests, audits monitoring of work performance, and materials analysis. The records shall also include closely-related data such as qualifications of personnel, procedures and equipment. Inspections and test records shall, as a minimum, identify the inspector or data recorder, the type of observation, the results, the acceptability, and the action taken in connection with any deficiency noted. Records shall be identifiable and retrievable. Consistent with applicable regulatory requirements, the applicant shall establish requirements concerning record retention, such as duration, location, and assigned responsibility."

- 10 CFR 71, Subpart H, Quality Assurance

- 10 CFR 72, Subpart D, Records, Reports, Inspections, and Enforcement

- 10 CFR 72, Subpart G, Quality Assurance

- 10 CFR 72.72, Inventory and Record Requirements for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Facility

- 10 CFR 72.174, Quality Assurance Records

Use the following regulatory guides in developing the recordkeeping requirements for the repository:

NOTE: Guidance identified in this section involves quality assurance references other than the QARD manual (DOE 1998). The QARD is the governing quality assurance

document for the Yucca Mountain Site Characterization Project (see Chapter 14), but the guidance identified here provides more specific information than that provided in the QARD to assist the author in preparing this section.

- Regulatory Guide 1.28, *Quality Assurance Program Requirements (Design and Construction)* (NRC 1985), describes a method acceptable to NRC staff for complying with the provisions of 10 CFR 50, Appendix B, with regard to establishing and implementing the requisite quality assurance program for the design and construction of nuclear power plants. It specifies those quality activities during design and construction for which procedural control is required, including recordkeeping. These activities are generally applicable to an MGR.

- Regulatory Guide 1.33, *Quality Assurance Program Requirements (Operation)* (NRC 1978e), describes a method acceptable to NRC staff for complying with NRC regulations with regard to overall quality assurance program requirements for the operation phase of nuclear power plants. This guide provides guidance to ensure minimum procedural coverage for plant operating activities, including maintenance activities that should be covered by written procedures. These procedures are generally applicable to spent fuel and high-level radioactive waste handling activities and many will require records retention.

- Regulatory Guide 3.48, *Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage)* (NRC 1986c), describes a standard format acceptable to NRC staff for the submittal of a Safety Analysis Report during License Application for these types of installations. Section 9.4, Normal Operations, and Section 9.4.2, Records, require that the applicant present a detailed management system for maintaining records relating to the historical operation of the installation. This system should include operating records, including principal maintenance, alterations, or additions made; records of off-normal occurrences and events associated with radioactive releases; environmental survey records; and the identity and pertinent information of the spent nuclear fuel or high-level radioactive waste stored.

- Regulatory Guide 3.50, *Standard Format and Content for a License Application to Store Spent Fuel and High-Level Radioactive Waste* (NRC 1989), requires in Chapter 8, Inventory and Records Systems, that the applicant provide a description of the inventory and records system for the stored spent nuclear fuel and high-level radioactive waste.

NOTE: Although the following draft regulatory guides are included here for consideration, they will be evaluated for applicability when they are formally issued as regulatory guides.

- DG-1010, *Quality Assurance Program Requirements (Design, Construction, Modification and Decommissioning)*, proposed Rev. 4 to Regulatory Guide 1.28 (NRC 1985), extends the scope of Regulatory Guide 1.28 to apply to modifications and decommissioning.

- DG-RS 902-4, *Quality Assurance Program Requirements (Operation)*, second proposed Rev. 3 to Regulatory Guide 1.33 (NRC 1978e), contains revised guidance affecting quality assurance programs initiated by revision to ANSI N18.7-1976/ANS 3.2-1994 (ANSI/ANS 1994) and changes in philosophy ("lessons learned") due to development by NRC and various other organizations.

- DG-1064, *An Approach for Plant-Specific, Risk-Informed Decision Making: Graded Quality Assurance*, provides potential changes to quality assurance requirements by incorporating the concept of graded quality assurance. This concept is intended to provide a safety (and cost) benefit by allowing NRC and licensees to preferentially allocate resources based on the safety significance of the item.

- DG- 3001, *Records Important for Decommissioning for Licenses Under 10 CFR Parts 30, 40, 70, and 72,* provides specific information that should be kept and maintained in the decommissioning records regarding the radiological conditions at the facility.

Describe the program for maintaining records of geologic repository activities to include a complete history of receipt, handling, storage, and disposition of radioactive wastes; construction records; reports of deficiencies of the site; records of permanent closure; records of tests using radioactive waste conducted during construction or emplacement; and records of inspections . State that, at a minimum, the following records shall be maintained:

- Repository construction records to include (10 CFR 60.72):
  - Surveys of the underground facility excavations, shafts, boreholes referenced to readily identifiable surface features or monuments
  - A description of the materials encountered
  - Geologic maps and geologic cross sections
  - Locations and amount of seepage
  - Details of equipment, methods, progress, and sequence of work
  - Construction problems
  - Anomalous conditions encountered
  - Instrument locations, readings, and analysis
  - Location and description of structural support systems
  - Location and description of dewatering systems
  - Details, methods of emplacement, and location of seals used.

- Records of facility changes, procedure changes, and tests and experiments performed pursuant to 10 CFR 60.44. These records shall include a safety evaluation providing the basis that the change, test, or experiment does not constitute an unreviewed safety question (10 CFR 60.44).

- Records of the receipt, handling, and disposition of radioactive waste. The records shall contain sufficient information to provide a complete history of the movement of the waste from the shipper through all phases of storage and disposal (10 CFR 60.7 1 (b)).

- Records and reports as required by 10 CFR 20. These records shall be maintained in accordance with the guidelines of Regulatory Guide 8.7, *Instructions for Recording and Reporting Occupational Radiation Exposure Data* (NRC 1992b), and Regulatory Guide 10.1, *Compilation of Reporting Requirements for Persons Subject to NRC Regulations* (NRC 1981d).

- Records important to facility decommissioning in accordance with guidelines of Regulatory Guide DG-3001, *Records Important For Decommissioning for Licenses Under 10 CFR Part 30, 40, 70, and 72.*

- Records of tests using radioactive waste conducted during construction or emplacement.

- Records of permanent closure.

- Records of inspections (10 CFR 75).

- Records of personnel training, qualification, and certification. Refer to Section 11.8.6 for details.

- Other records as required by the conditions of the authorization to construct or to receive and possess radioactive waste.

11.4.2 Record Retention Periods

Regulatory requirements and guidance concerned with initiation and retention of records include, but are not limited to, the following:

- 10 CFR 60, Disposal of High-Level Radioactive Wastes in Geologic Repositories, which reads in part:

  - Section 60.2, Definitions: "Unanticipated processes and events means...(5) relevant records are preserved, and remain accessible, for several hundred years after permanent closure."

  - Section 60.4, Communications and Records: "Each record required by this part must be legible throughout the retention period..."

- Section 60.51, License Amendment for Permanent Closure: "...(2)(ii) Placement of records in the archives and land record systems of Local, State and Federal Government agencies, and archives elsewhere in the world that would be likely to be consulted by potential human intruders..."

- Section 60.72, Construction Records: "(a) The DOE shall maintain records of construction of the geologic repository operations area in a manner that ensures their usability for future generations..."

The following regulatory guides shall be used in developing retention periods of required quality records:

NOTE: Guidance identified in this section involves quality assurance references other than the QARD manual (DOE 1998). The QARD is the governing quality assurance document for the Yucca Mountain Site Characterization Project (see Chapter 14), but the guidance identified here provides more specific information than that provided in the QARD to assist the author in preparing this section.

- Regulatory Guide 1.28, *Quality Assurance Program Requirements (Design and Construction)* (NRC 1985), describes a method acceptable to NRC staff for complying with the provisions of 10 CFR 50, Appendix B, with regard to establishing and implementing the requisite quality assurance program for the design and construction of nuclear power plants. It specifies those quality activities during design and construction for which procedural control is required, including recordkeeping. Section 2.8 addresses specifically the retention of non-permanent quality assurance records.

- Regulatory Guide 3.48, *Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage)* (NRC 1986c), describes a standard format acceptable to NRC for the submittal of a Safety Analysis Report during License Application for these types of installations. Sections 9.4 and 9.6 require that the applicant describe plans to obtain and safeguard records and archive files that will support decommissioning.

NOTE: Although the following draft regulatory guide is included here for consideration, it will be evaluated for applicability when formally issued as a regulatory guide.

DG-3001, *Records Important for Decommissioning for Licenses Under 10 CFR Parts 30, 40, 70, and 72,* provides specific information that should be kept and maintained in the decommissioning records regarding the radiological conditions at the facility.

Discuss record retention times and 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. Guidance for facility records management systems is provided in Regulatory Guide 1.28 (NRC 1985), ANSI N18.7-1976/ANS 3.2-1994 (ANSI/ANS 1994), and ASME-NQA-1-1989 (ASME 1989). The applicability of this guidance will be determined, and

appropriate instructions for the License Application author will be provided, in a future revision to this technical guidance document. Commercial nuclear utility records control systems should serve as a guide to meet NRC expectations and resolve concerns that the records program is sufficient.

11.5 PROPOSED OPERATING CONTROLS AND LIMITS

Identify the variable conditions or other items that are proposed to be the subject of repository operating controls or limits. Justify the selection of these items by providing the bases for inclusion. Include applicable design criteria or the results of repository safety analyses.

The operating limits and controls must be identified to the fullest extent possible, including numerical values and other pertinent data. Also identify the applicable sections of the Safety Analysis Report that develop the details and bases of the control or limit. In summary, the operating controls and limits must be derived from facility safety analysis and include all aspects of repository operations that are important to safety, retrievability, and waste isolation. Operation of the repository within the proposed operating controls and limits must ensure the health and safety of the worker, the public, and the environment. Format for the repository operating controls and limits is discussed below.

The suggested format for operating controls and limits is as follows:

- Title: Write the name or description of the limit or control, such as maximum dose rate at any level of a nuclear fuel storage structure.
- Specification: Identify the limits.
- Applicability: Clearly define systems or operations to which the control or limit applies.
- Objective: State reasons for the control or limit and the specific unsafe conditions it is intended to prevent.
- Action: Clearly 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: The Safety Analysis Report must contain all pertinent information and an explicit detailed analysis and assessment supporting the choice of the item and its specific value or characteristics. The bases for each control or limit 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. The following subjects are appropriate for discussion in the bases section:
  - Technical Basis: The technical basis is derived from technical knowledge of the process and its characteristics and must support the choice of the particular

variable as well as the value of the variable. Summarize the results of computations, experiments, or judgments, and analysis and evaluation.

- Equipment: A safety limit often is protected by or closely related to certain equipment. Note any such relationship, and state the means by which the variable is monitored and controlled. Note the function of the equipment and how and why the requirement is selected. In addition, note the means by which surveillance is accomplished. If periodic surveillance is required, give the basis for frequency of required action.

- Operation: State the margins and the bases that relate to the safety limits and the normal operating zones. State the roles of operating procedures and of protective systems in guarding against exceeding a limit or condition. Include a brief discussion of such factors as system responses, process or operational transients, malfunctions, and procedural errors. Make reference to related controls or limits.

11.6 PRESERVATION OF RECORDS

Describe the system or process that is established for the permanent preservation of site records, including site data prior to site characterization, data from site characterization experiments and in situ tests, construction records, and performance confirmation records. Show how the system meets the requirements of 10 CFR 60.51(a)(2)(ii).

ANSI/NFPA 232-1995, *Protection of Records* (ANSI/NFPA 1995), provides standards for the protection of records and information. The National Archives and Records Administration also publishes guidance concerning records management, archiving, and preservation, including the National Records and Information Management Handbook, *Evaluating a Vital Records Program* (National Archives and Records Administration 1992a), and *Preservation of Archival Records: Holdings Maintenance at the National Archives* (National Archives and Records Administration 1992b). The applicability of this guidance will be determined, and appropriate instructions for the License Application author will be provided, in a future revision to the technical guidance document.

11.7 SITE MARKERS

Describe the system of site markers (including monuments) used to permanently identify the repository following permanent closure and to prevent future human interactions that may reduce the ability of the repository to isolate wastes. State that the controlled area and geologic repository operations area will be identified by markers that have been designed, fabricated, and emplaced to be as permanent as practicable. Introduce this section by stating the objective of the site marker system and summarizing the section information. State that these site markers are designed to minimize the potential for intrusion that has the potential to result in releases of radioactive material to the accessible environment.

State that the repository site marker system is in compliance with the requirements of 10 CFR 60.21(c)(8) and 10 CFR 60.51(a)(2)(i). These site markers, in conjunction with the provisions for

record preservation of Section 11.6, satisfy the regulatory requirements to minimize the potential for human intrusion into the repository, to restrict access, and to avoid disturbances that could breach the barrier system for at least 1,000 years.

11.7.1 General Site Description

Provide a general description of the repository site from the perspective of placement of site markers, including a discussion of outstanding demographic and geologic features of the site and vicinity. Provide a scale drawing depicting the locations of the geologic repository operations area and controlled area, permanent structures, engineered barriers, site roads, and all natural and man-made barriers. Show the location of proposed site markers with respect to the features described above. Discuss the future changes in site geomorphology considered in placement of the markers.

11.7.2 Site Marker Description

Describe the design of the proposed site markers. Include dimensions, materials, construction, and markings or symbols. Discuss the permanence of the markers and the features that serve to convey a clear warning of the dangers of the underlying hazard. Discuss the performance of the site markers, including the bases for design service life.

Indicate that the design basis for the site markers is to provide a local indication that the site presents unique hazards, and that the area should not be disturbed. The form and characteristics of government and civilization are difficult to project over the length of time the repository must safely contain the spent nuclear fuel and high-level radioactive waste. Therefore, the site markers are also intended to indicate that further research is necessary before disturbing the site in the case where the marker warnings themselves are not understood.

11.7.3 Additional Access Control Methods

Discuss any additional methods, passive or active, that may be used in the postclosure period to restrict access and regulate land use at the site and adjacent areas to prevent adverse human actions that could significantly reduce the geologic repository's ability to achieve isolation.

11.8 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 10 CFR 60, Subpart H, Training and Certification of Personnel, for issuance of an NRC license. Specifically, this section will meet the requirements of 10 CFR 60.160, General Requirements; 10 CFR 60.161, Training and Certification Program; and 10 CFR 60.162, Physical Requirements.

This section develops, documents, and implements a training program with a goal 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 non-safety related and as important to safety in the safety analysis and the license.

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

11.8.1 Training Organization

This section includes:

- A basic training department organizational chart
- A description of the training department personnel functions, responsibilities, and authorities
- The training department organizational goals and objectives

State that it is the responsibility of line management to ensure their personnel are trained and qualified to safely and effectively perform job requirements.

State that the training department will develop training programs using a systematic approach to training to ensure the qualification of personnel in accordance with 10 CFR 60, Subpart H, requirements.

The following documents contain training commitments used to identify training program components for the development of training programs for the performance of tasks identified by an MGR job and task analysis:

- ANSI/ANS-3.1-1993 (ANSI/ANS 1993), *Selection, Qualification, and Training of Personnel for Nuclear Power Plants*, Section 6, Training
- Regulatory Guide 1.70, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants* (NRC 1978f), Section 13.2, Training

11.8.2 Development and Qualification of Training Staff

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. 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.

ANSI/ANS-3.1-1993 (ANSI/ANS 1993), *Selection, Qualification, and Training of Personnel for Nuclear Power Plants*, Section 4.5.4, Instructors, contains training commitments used to identify

training program components for the qualification of instructors with full-time duties and responsibilities for instructing personnel or developing training materials.

11.8.3 Personnel Selection and Entry-Level Requirements

Describe the procedures or policies defining personnel selection and entry-level requirements for facility personnel. Entry-level requirements will specify the minimum education, technical, and experience requirements necessary for an employee to meet the job requirements.

Establish that facility management is responsible for only hiring or transferring 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.

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 (10 CFR 60.162).

The following documents contain training commitments used to identify training program components for the selection of personnel for the performance of tasks identified by an MGR job and task analysis:

- ANSI/ANS-3.1-1993 (ANSI/ANS 1993), *Selection, Qualification, and Training of Personnel for Nuclear Power Plants*, Section 6, Training
- Regulatory Guide 1.8, *Qualification and Training of Personnel for Nuclear Power Plants* (NRC 1987a)

11.8.4 Facility Personnel Training

This section describes the non-safety-related facility training program. This program is based on a systematic approach to training, as well as the established hazards of the facility. Clearly 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 are implemented to ensure facility personnel are qualified to perform job requirements in a safe and efficient manner. Describe programs for general employee, technician and maintenance (non-safety-related), health physics, emergency response, technical staff, and management and supervisory training. Also include other training programs specific to the repository, such as underground safety training.

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 performance of trained facility personnel.

The following documents contain training commitments used to identify training program components for the development of training programs for facility personnel training (non-safety-related):

- Regulatory Guide 8.27, *Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants* (NRC 1981c)
- Regulatory Guide 8.8, *Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Practicable* (NRC 1978a), Section C.1.c, Training and Instruction
- Regulatory Guide 8.10, *Operating Philosophy for Maintaining Occupational Radiation Exposure As Low As Reasonably Achievable* (NRC 1975a)
- Regulatory Guide 8.13, *Instruction Concerning Prenatal Radiation Exposure* (NRC 1981b)
- Regulatory Guide 8.15, *Acceptable Programs for Respiratory Protection* (NRC 1976a)
- Regulatory Guide 8.29, *Instruction Concerning Risks from Occupational Radiation Exposure* (NRC 1996a)
- ASME B30.2-1996, *Overhead and Gantry Cranes* (ASME 1996), Section 2-3.1, Qualifications for Conduct of Operators
- NUREG-0612, *Control of Heavy Loads at Nuclear Power Plants* (NRC 1980a), Section SRP 13.1.3, Qualifications or Nuclear Plant Personnel

11.8.5 Qualification and Certification Process

Describe the program for the qualification and certification of facility operators, supervisors, and maintenance personnel performing operations of system and components import to safety including requalification training and exceptions to training requirements. State that only these certified personnel or other personnel under the direct visual supervision of a certified individual shall operate system and components important to safety (10 CFR 60.160 and 10 CFR 161). Also discuss testing requirements to determine the qualifications for facility operators,

supervisors, and maintenance personnel. State that the qualification and certification training program will be comprised of 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

The following documents contain training commitments used to identify training program components for the development of training programs for personnel requiring qualification and certification (safety-related):

- 10 CFR 55, Operators' Licenses (this reference will be used for general qualification, certification, and requalification program requirements as they are applicable to the MGR)
- Regulatory Guide 1.8, *Qualification and Training of Personnel for Nuclear Power Plants* (NRC 1987a)
- Regulatory Guide 3.68, *Nuclear Criticality Safety Training* (NRC 1993)
- ANSI/ANS-3.1-1993, *Selection, Qualification, and Training of Personnel for Nuclear Power Plants* (ANSI/ANS 1993)
- Regulatory Guide 1.70, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants LWR Edition* (NRC 1978f)
  - Section 13.1.1.3, Qualifications
  - Section 13.1.3, Qualifications of Nuclear Plant Personnel
  - Section 13.2, Training
- Regulatory Guide HF 608-4, *Training and Certification of Independent Spent Fuel Storage Installation Operators*

11.8.6 Training Records

Describe the administrative program for the maintenance of training, qualification, and certification records for MGR organization personnel. Training records will be maintained to allow the review of course content, schedule, trainee attendance, evaluation results, and employee qualification.

The following documents contain training commitments used to identify training program components for the maintenance of training records:

- TG17-1993, *Management of Nuclear Related Training Records* (NIRMA 1993)
- Privacy Act of 1974 for privileged records

11.9 EMERGENCY PLANNING

NRC will review the emergency planning information as described in a separately submitted document containing the emergency plan and as summarized in this section of the License Application. The emergency planning information should provide evidence of preliminary planning for emergency preparedness directed at situations involving real or potential radiological hazards.

Currently 10 CFR 60, Subpart I, Emergency Planning Criteria, is reserved. Until promulgated, no federal guidance exists specific to the MGR. It is likely, however, MGR criteria and NRC review will be similar to the planning standards described in 10 CFR 50.47(b), 10 CFR 30.32, 10 CFR 70.22, and 10 CFR 72.24(k).

Demonstrate that the emergency planning is discussed in accordance with the regulatory guides listed below, as appropriate to the topical area under consideration.

Regulatory Guidance

The following regulatory guidance applies to emergency planning:

- Regulatory Guide 3.67, *Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities* (NRC 1992c), provides a detailed outline, categorized under topical headings, of the information NRC expects to be included in emergency plans prepared for fuel cycle and materials facilities. The format and content of the emergency plan shall be consistent with the format and content guidance provided in this guide. Modifications to the content and format as needed to adequately address site specific characteristics are acceptable.

- Regulatory Guide 3.42, *Emergency Planning for Fuel Cycle Facilities and Plants Licensed Under 10 CFR Parts 50 and 70* (NRC 1979b), includes significant detail on the information NRC expects to be considered during development of an emergency plan. Although the recommended format found in this guide is not consistent with that recommended in Regulatory Guide 3.67 (NRC 1992c), the level of detail in the discussions will provide a useful resource for preparation of the emergency plan and shall be considered to be applicable from the content (not format) perspective.

- Regulatory Guide 1.101, *Emergency Planning and Preparedness for Nuclear Power Reactors* (NRC 1992a), is specific to reactor licensees subject to 10 CFR 50.47 (Emergency Plans) and was prepared specifically to cover the incorporation of Federal Emergency Management Agency criteria in NRC reactor licensee emergency

plans. The referenced document upon which the regulatory guide is based is NUREG-0654/FEMA-REP-1 (NRC 1988), which is considered by NRC to provide an acceptable method for complying with 10 CFR 50.47. Additionally, this guide cites NUMARC/NESP-007 (Methodology for Development of Emergency Action Levels) as an acceptable alternative to a portion of NUREG-0654 (NRC 1988). Although this guide is applicable to reactors, its provisions relative to incorporating Federal Emergency Management Agency/NRC agreements shall be considered for the repository emergency plan.

- NUREG-0654/FEMA-REP-1, *Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants* (NRC 1988)

- NUREG-0696, *Functional Criteria for Emergency Response Plans and Preparedness in Support of Nuclear Power Plants* (NRC 1981a)

The geologic repository emergency plan, as required by 10 CFR 60.21(c)(9), will be submitted to NRC as a separate document to accompany the License Application. This section of the License Application, which is a part of the Safety Analysis Report, should reference the emergency plan and present an overview of its contents. The information presented should provide evidence of preliminary planning. This summary, as described for a monitored retrievable storage installation in the *Standard Review Plan for Spent Fuel Dry Storage Facilities*, NUREG-1567 (NRC 1996b), can be limited to the following information:

- A statement that the outline and content of the emergency plan submitted with the License Application is in accordance with the applicable regulatory requirements. Specify the applicable regulatory requirements (e.g., "under 10 CFR 72.32(b)").

- A concise summary of the emergency plan, including a brief description of each section may be incorporated.

- A listing and brief description of the types of radioactive material accidents provided for by the emergency plan. All accident-level events or conditions addressed in the accident analysis or design basis event section(s) should be included.

- For construction authorization, present a preliminary list of the types of offsite response organizations that will be provided with an opportunity to comment on the plan. Prior to the first full scale onsite or offsite graded exercise, present the identities of offsite response organizations provided opportunity to comment on the plan, and a summary of responses in accordance with the applicable regulatory requirements. Specify the applicable regulatory requirements (e.g., "under 10 CFR 72.32(b)(14)").

- For construction authorization, provide a preliminary list of the types of organizations expected to provide offsite assistance. Prior to the first full scale onsite or offsite

graded exercise, present the identities of organizations with whom arrangements have been made for offsite assistance in accordance with the applicable regulatory requirements. Specify the applicable regulatory requirements.

Emphasize in greater detail those practices or features of emergency planning for the geologic repository that are substantially different from elsewhere in the nuclear industry. It is important to provide in the overview section of the Safety Analysis Report a sufficient level of detail to give the NRC reviewers confidence that the project will implement the emergency planning program that will meet or exceed requirements.

11.10 REQUIREMENTS FOR ADDITIONAL TECHNICAL INFORMATION

The License Application must be as complete as possible using information that is reasonably available at the time of submission. Provide an explanation, with reference to information not supplied at the time of submission, of why such information was not reasonably available.

As required by 10 CFR 60.21(c)(14), identify, describe, and discuss those safety features or components for which further technical information is required in support of the issuance of a license, but has not been supplied in the submittal of the License Application. This includes items required as a part of the performance confirmation program to support predictions of postclosure performance, in addition to other items unrelated to this program. Those items of additional information required as a part of the performance confirmation program should be listed here, with a cross-reference to detailed discussion of this topic in Chapter 12.

Include the following information in this section:

- Identify and distinguish between those technical information development programs that will be required to determine the adequacy of a new design and those that will be used to demonstrate the margin of conservatism of a proven design.
- 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 License Application 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.

References may be made to topical program or issue resolution summary reports filed with NRC; however, if such references are made, the applicability of each technical information development item to the repository should be discussed.

Include a description of any special technical information development programs undertaken to establish the final design and/or demonstrate the conservatism of the design, and a discussion of any programs that will be conducted during operation in order to demonstrate the acceptability of contemplated future changes in design or operation.

11.11 PROPOSED TECHNICAL SPECIFICATIONS

Demonstrate compliance with the requirements stated in 10 CFR 60.21(c)(6), 10 CFR 60.42(a), and 10 CFR 60.43. State that the purpose of the technical specifications is to provide the variables, conditions, or other items that govern the licensed operation of the facility. The technical 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. NRC will review DOE-proposed technical specifications, revise and augment them as deemed appropriate, and issue the final technical specifications as an NRC document.

V. REFERENCES

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10 CFR 55. Energy: Operators' Licenses.

10 CFR 60. Energy: Disposal of High-Level Radioactive Wastes in Geologic Repositories.

10 CFR 70. Energy: Domestic Licensing of Special Nuclear Material.

10 CFR 71. Energy: Packaging and Transportation of Radioactive Material.

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CHAPTER 12. PERFORMANCE CONFIRMATION PROGRAM

I. OBJECTIVE

This chapter provides guidance for the author(s) of Chapter 12 of the License Application. The License Application will be submitted to the U.S. Nuclear Regulatory Commission (NRC) for authorization to construct a high-level radioactive waste repository at Yucca Mountain. This chapter must describe the performance confirmation program in sufficient detail to demonstrate that an adequate program has been implemented to ensure that the natural and engineered systems and components required for repository operation and important to waste isolation, or which are designed or assumed to operate as barriers after permanent closure, are functioning as intended and anticipated.

Performance confirmation, as defined in 10 CFR 60.2, means the program of tests, experiments, and analyses which is conducted to evaluate the accuracy and adequacy of the information used to determine with reasonable assurance that the performance objectives for the period after permanent closure will be met.

II. GUIDANCE TOPICS

The following topics will be discussed in Chapter 12 of the License Application:

- 12.0 PURPOSE, SUMMARY, AND CONCLUSIONS
- 12.1 PERFORMANCE CONFIRMATION PLAN
- 12.2 PERFORMANCE CONFIRMATION FOR THE NATURAL BARRIERS
  - 12.2.1 Geologic System
  - 12.2.2 Hydrologic System
  - 12.2.3 Geochemical System
  - 12.2.4 Climatological and Meteorological Systems
- 12.3 PERFORMANCE CONFIRMATION FOR THE SYSTEMS, STRUCTURES, AND COMPONENTS OF THE GEOLOGIC REPOSITORY OPERATIONS AREA
  - 12.3.1 Surface Facilities
  - 12.3.2 Shafts and Ramps
  - 12.3.3 Underground Facility
- 12.4 PERFORMANCE CONFIRMATION FOR THE WASTE PACKAGE AND ENGINEERED BARRIER SYSTEM
  - 12.4.1 Waste Package Monitoring
  - 12.4.2 Engineered Barrier and Waste Package Performance Objectives
- 12.5 RADIATION PROTECTION FOR PERFORMANCE CONFIRMATION ACTIVITIES
- 12.6 ANALYSIS OF CHANGES FROM PERFORMANCE CONFIRMATION BASELINE
- 12.7 UNRESOLVED SAFETY QUESTIONS

III. REGULATORY REQUIREMENTS

***** IMPORTANT NOTE *****

At the time of development of this document, it appears likely that NRC will issue a new, site-specific regulation for Yucca Mountain in 1999. At the time the new 10 CFR 63 is issued, 10 CFR 60 will be modified to state that it is not applicable to Yucca Mountain. Details about the new regulation are not known. However, it has been determined prudent to develop the working draft License Application in accordance with the best available estimate of the form of the new regulation. To avoid confusing references and to allow production of the technical guidance document in time to support training for the working draft License Application, the text of the technical guidance document that follows this note has not been revised to reflect the postulated form of the new regulation. Instead, this note provides guidance needed by working draft License Application authors to comply with management direction on addressing the new regulation. The paragraphs that follow explain changes in the regulation with respect to the existing 10 CFR 60 that affect this technical guidance document chapter and provide guidance on how to address the changes in the working draft License Application. Where guidance in this note conflicts with guidance in the remainder of this chapter of the technical guidance document, the guidance in the note is to govern. Any questions should be addressed to Management and Operating Contractor (M&O) Licensing.

Changes to Regulations Applicable to Chapter 12

The following 10 CFR 60 requirements applicable to technical guidance document Chapter 12 and listed below in Section III are assumed to not be carried over to the new regulation: 60.130, 60.131, and 60.136.

In place of these specific requirements, the new regulation is expected to require a discussion of the contribution of the individual barriers to the overall performance of the repository. The manner in which that contribution is to be described has not been specified, though an NRC review plan on the subject may be issued sometime in the future.

Guidance to Working Draft License Application Authors for Addressing Changes

- Disregard guidance in Section IV regarding demonstrating compliance with any of the regulations assumed to be deleted as described above. Instead, discuss the conduct of the performance confirmation program in supporting overall repository performance.

- Address the technical issue of each of the regulations in Section III, including those expected to be deleted, but do not relate the discussion to demonstrating compliance with any of the deleted regulations. Continue to explicitly address the regulations in Section III that are not expected to be deleted.

- Detailed discussion of design criteria, design bases, and performance is still required as per the existing guidance in Section IV. Discussion should still include detailed discussions of the phenomena that affect that performance.

- No sections or subsections of chapter 12 are expected to be deleted as a result of this note.

***** END OF NOTE *****

Sections of the regulations that are to be addressed fully or in part in Chapter 12 of the License Application are reproduced below. These are verbatim extracts from the regulations. Paragraph numbers refer to Title 10 of the U.S. Code of Federal Regulations (CFR). In addition, Appendix A provides a cross-reference between the regulations in 10 CFR 60 and the specific License Application sections that address these regulations. Revisions of regulations are not uncommon. Verify that current versions are used.

10 CFR Part 60, Disposal of High-Level Radioactive Wastes in Geologic Repositories

60.21 Content of application.

(c) The Safety Analysis Report shall include:

(1) A description and assessment of the site at which the proposed geologic repository operations area is to be located with appropriate attention to those features of the site that might affect geologic repository operations area design and performance...

(ii) The assessment shall contain:

(E) An analysis of the performance of the major design systems, structures, and components, both surface and subsurface, to identify those that are important to safety. For the purposes of this analysis, it shall be assumed that operations at the geologic repository operations area will be carried out at the maximum capacity and rate of receipt of radioactive waste stated in the application...

(2) A description and discussion of the design, both surface and subsurface, of the geologic repository operations area including: (i) the principal design criteria and their relationship to any general performance objectives promulgated by the Commission, (ii) the design bases and the relation of the design bases to the principal design criteria, (iii) information relative to materials of construction (including geologic media, general arrangement, and approximate dimensions), and (iv) codes and standards that the DOE proposes to apply to the design and construction of the geologic repository operations area...

(3) A description and analysis of the design and performance requirements for systems, structures, and components of the geologic repository which are important to safety. The analysis must include a demonstration that--(i) The requirements of §

60.111(a) will be met, assuming occurrence of Category 1 design basis events; and (ii) The requirements of § 60.136 will be met, assuming occurrence of Category 2 design basis events.

(14) An identification of those systems, structures, and components of the geologic repository, both surface and subsurface, which require research and development to confirm the adequacy of design. For systems, structures, and components important to safety and for the engineered and natural barriers important to waste isolation, the DOE shall provide a detailed description of the programs designed to resolve safety questions, including a schedule indicating when these questions would be resolved.

60.111 Performance of the geologic repository operations area through permanent closure.

(a) *Protection against radiation exposures and releases of radioactive material.* The geologic repository operations area shall be designed so that until permanent closure has been completed, radiation exposures and radiation levels, and releases of radioactive materials to unrestricted areas, will at all times be maintained within the limits specified in Part 20 of this chapter and such generally applicable environmental standards for radioactivity as may have been established by the Environmental Protection Agency.

60.130 Scope of design criteria for the geologic repository operations area.

Pursuant to the provisions of § 60.21(c)(2)(i), an application to receive, possess, store, and dispose of high-level radioactive waste in the geologic repository operations area must include the principal design criteria for a proposed facility. The principal design criteria establish the necessary design, fabrication, construction, testing, maintenance, and performance requirements for structures, systems, and components important to safety and/or important to waste isolation. Subsections 60.131 through 60.134 specify minimum requirements for the principal design criteria for the geologic repository operations area. These design criteria are not intended to be exhaustive. However, omissions in §§ 60.131 through 60.134 do not relieve DOE from any obligation to provide such features in a specific facility needed to achieve the performance objectives.

60.131 General design criteria for the geologic repository operations area.

(a) *Radiological protection.* The geologic repository operations area shall be designed to maintain radiation doses, levels, and concentrations of radioactive material in air in restricted

areas within the limits specified in Part 20 of this chapter. Design shall include:

(1) Means to limit concentrations of radioactive material in air;

(2) Means to limit the time required to perform work in the vicinity of radioactive materials, including, as appropriate, designing equipment for ease of repair and replacement and providing adequate space for ease of operation;

(3) Suitable shielding;

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

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

(6) A radiation alarm system to warn of significant increases in radiation levels, concentrations of radioactive material in air, and of increased radioactivity released in effluents. The alarm system shall be designed with provisions for calibration and for testing its operability.

(b) *Protection against design basis events.* The structures, systems, and components important to safety shall be designed so that they will perform their necessary safety functions, assuming occurrence of design basis events.

(c) *Protection against dynamic effects of equipment failure and similar events.* The structures, systems, and components important to safety shall be designed to withstand dynamic effects such as missile impacts, that could result from equipment failure, and similar events and conditions that could lead to loss of their safety functions.

(d) *Protection against fires and explosions.* (1) The structures, systems, and components important to safety shall be designed to perform their safety functions during and after credible fires or explosions in the geologic repository operations area.

(2) To the extent practicable, the geologic repository operations area shall be designed to incorporate the use of noncombustible and heat resistant materials...

(4) The geologic repository operations area shall be designed to include means to protect structures, systems, and components important to safety against the adverse effects of either the operation or failure of the fire suppression systems.

(e) *Emergency capability.* (1) The structures, systems, and components important to safety shall be designed to maintain control of radioactive waste and radioactive effluents, and permit prompt termination of operations and evacuation of personnel during an emergency...

(g) *Inspection, testing, and maintenance.* The structures, systems, and components important to safety shall be designed to

permit periodic inspection, testing, and maintenance, as necessary, to ensure their continued functioning and readiness...

(i) *Instrumentation and control systems.* The design shall include provisions for instrumentation and control systems to monitor and control the behavior of systems important to safety, assuming occurrence of design basis events.

Preclosure Controlled Area

60.136 Preclosure controlled area.

(b) The geologic repository operations area shall be designed so that, for Category 2 design basis events, no individual located on or beyond any point on the boundary of the preclosure controlled area will receive the more limiting of a total effective dose equivalent of 0.05 Sv (5 rem), or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.5 Sv (50 rem). The eye dose equivalent shall not exceed 0.15 Sv (15 rem), and the shallow dose equivalent to skin shall not exceed 0.5 Sv (50 rem). The minimum distance from the surface facilities in the geologic repository operations area to the boundary of the preclosure controlled area must be at least 100 meters.

Subpart F — Performance Confirmation Program

60.140 General requirements.

(a) The Performance Confirmation Program shall provide data which indicates, where practicable, whether:

(1) Actual subsurface conditions encountered and changes in those conditions during construction and waste emplacement operations are within the limits assumed in the licensing review; and

(2) Natural and engineered systems and components required for repository operation, or which are designed or assumed to operate as barriers after permanent closure, are functioning as intended and anticipated.

(b) The program shall have been started during site characterization and it will continue until permanent closure.

(c) The program shall include *in-situ* monitoring, laboratory and field testing, and *in-situ* experiments, as may be appropriate to accomplish the objective as stated above.

(d) The program shall be implemented so that:

(1) It does not adversely affect the ability of the natural and engineered elements of the geologic repository to meet the performance objectives.

(2) It provides baseline information and analysis of that information on those parameters and natural processes pertaining

to the geologic setting that may be changed by site characterization, construction, and operational activities.

(3) It monitors and analyzes changes from the baseline condition of parameters that could affect the performance of a geologic repository.

(4) It provides an established plan for feedback and analysis of data, and implementation of appropriate action.

60.141 Confirmation of geotechnical and design parameters.

(a) During repository construction and operation, a continuing program of surveillance, measurement, testing, and geologic mapping shall be conducted to ensure that geotechnical and design parameters are confirmed and to ensure that appropriate action is taken to inform the Commission of changes needed in design to accommodate actual field conditions encountered.

(b) Subsurface conditions shall be monitored and evaluated against design assumptions.

(c) As a minimum, measurements shall be made of rock deformations and displacement, changes in rock stress and strain, rate and location of water inflow into subsurface areas, changes in groundwater conditions, rock pore water pressures including those along fractures and joints, and the thermal and thermomechanical response of the rock mass as a result of development and operations of the geologic repository.

(d) These measurements and observations shall be compared with the original design bases and assumptions. If significant differences exist between the measurements and observations and the original design bases and assumptions, the need for modifications to the design or in construction methods shall be determined and these differences and the recommended changes reported to the Commission.

(e) *In-situ* monitoring of the thermomechanical response of the underground facility shall be conducted until permanent closure to ensure that the performance of the natural and engineering features are within design limits.

60.142 Design testing.

(a) During the early or developmental stages of construction, a program for *in situ* testing of such features as borehole and shaft seals, backfill, and the thermal interaction effects of the waste packages, backfill, rock, and groundwater shall be conducted.

(b) The testing shall be initiated as early as is practicable. Placement and compaction procedures against design requirements before permanent backfill placement is begun.

(d) Test sections shall be established to test the effectiveness of borehole and shaft seals before full-scale operation proceeds to seal boreholes and shafts.

60.143 Monitoring and testing waste packages.

(a) A program shall be established at the geologic repository operations area for monitoring the condition of the waste packages. Waste packages chosen for the program shall be representative of those to be emplaced in the underground facility.

(b) Consistent with safe operation at the geologic repository operations area, the environment of the waste packages selected for the waste package monitoring program shall be representative of the environment in which the wastes are to be emplaced.

(c) The waste package monitoring program shall include laboratory experiments which focus on the internal condition of the waste packages. To the extent practical, the environment experienced by the emplaced waste packages within the underground facility during the waste package monitoring program shall be duplicated in the laboratory experiments.

(d) The waste package monitoring program shall continue as long as practical up to the time of permanent closure.

IV. ACCEPTANCE CRITERIA AND GUIDANCE

General Guidance

There are no NRC guidance documents that provide detailed guidance on the development of a repository performance confirmation program. Limited NRC guidance can be obtained from NUREG-1323, *License Application Review Plan for a Geologic Repository for Spent Nuclear Fuel and High-Level Radioactive Waste* (NRC 1995), and Regulatory Guide DG-3003, *Format and Content for the License Application for the High-Level Waste Repository* (NRC 1990). The guidance in this chapter is, in general, based on these two documents and on the *Performance Confirmation Plan* (CRWMSM&O 1997). NUREG-1323 (NRC 1995) is largely incomplete and will become inapplicable to Yucca Mountain upon the expected issuance of a new 10 CFR 63 as discussed in Section III above. Information from NUREG-1323 (NRC 1995) considered likely to be applicable under the new regulation is included in the technical guidance document, though the NUREG is not referenced as an authoritative source for such information. As is the case for NUREG-1323 (NRC 1995), DG-3003 (NRC 1990) will become inapplicable to Yucca Mountain upon the expected issuance of a new 10 CFR 63 as discussed in Section III above. Information from DG-3003 (NRC 1990) considered likely to be applicable under the new regulation is included in the technical guidance document, though DG-3003 is not referenced as an authoritative source for such information. The performance confirmation program begins during site characterization and continues through permanent closure of the repository. The License Application will bring to closure the performance confirmation baseline in the *Site Characterization Plan Yucca Mountain Site, Nevada Research and Development Area, Nevada* (DOE 1988) and explain how this baseline will be carried over into the repository operation

performance confirmation program. Initial selection of performance confirmation information to be obtained as part of this program will be the subject of negotiation with NRC.

The preliminary nature of some information to be provided in the License Application will heavily influence development of the performance confirmation program. Ongoing performance confirmation activities will result in refinements to the information submitted in the License Application or revisions to the program. In addition, the performance confirmation program that is described in the License Application will be reviewed by the NRC staff. Changes to the program may result from this review and from NRC needs relative to a finding of reasonable assurance. Such changes will appear in subsequent License Applications, as appropriate.

Chapter 12 of the License Application relies heavily on information presented in other chapters. Summaries from other chapters (the need for which is stated in the sections that follow) should be presented in a tabular format, whenever possible. The information in this chapter should be clear and should not require the reviewer to conduct extensive searches for information contained in referenced chapters elsewhere in the License Application. The bases or justification behind the information used from other chapters generally need not be repeated in this chapter, as long as appropriate reference to that supporting information is made in the chapter.

In appropriate locations in this chapter, for any computer codes discussed, identify the computer code and the verification and validation status of the computer code. Identify data which require qualification. Identify the extent of compliance of any peer review and expert elicitation results with the requirements of DOE-RW/0333P, *Quality Assurance Requirements and Description* (DOE 1998).

12.0 PURPOSE, SUMMARY, AND CONCLUSIONS

In this chapter of the License Application, describe the performance confirmation program that will provide data to demonstrate the natural and engineered systems and components required for repository operation (or those which are designed or assumed to operate as barriers after permanent closure) are functioning as intended and anticipated to meet the requirements of 10 CFR 60, Subpart F, as required by 10 CFR 60.137. Performance confirmation, as defined in 10 CFR 60.2, means the program of tests, experiments, and analyses which is conducted to evaluate the accuracy and adequacy of the information used to determine with reasonable assurance that the performance objectives for the period after permanent closure will be met.

The performance confirmation program includes those activities that appear to be essential to reduce uncertainty in performance. It includes obtaining additional data, further developing process models, and refining numerical models. Its results will be used to support the NRC findings of reasonable assurance at the time of the License Application to receive and possess radioactive materials. Discuss these aspects of the program.

Present a summary of the chapter information, including:

- Major parameters to be monitored and tests to be performed

- Use of monitoring and test results conducted prior to closure to evaluate or confirm repository performance objectives for the period after permanent closure

- Duration/schedule of performance confirmation activities

State conclusions regarding the adequacy of the performance confirmation program. Present summary information that supports the conclusions that the performance confirmation program:

- Does not adversely affect the ability of the natural and engineered elements of the geologic repository to meet any of the 10 CFR 60 postclosure performance objectives

- Provides sufficient baseline information and an analysis of that information on those parameters and natural processes pertaining to the geologic setting that may be changed by site characterization, construction, and operational activities

- Monitors and analyzes changes from the baseline condition of parameters that could affect the postclosure performance of a geologic repository

- Provides an established plan for feedback and analysis of data, and implementation of appropriate action, including plans to ensure that appropriate and timely action is taken to inform NRC of changes in the field conditions being monitored and the subsequent need for changes in design, construction method, or a 10 CFR 60 postclosure performance objective

Identify the subissues of the NRC key technical issues (KTIs) that are addressed in this chapter. Refer to the more detailed discussions in identified sections/subsections that follow for information regarding status of the subissues and for discussion of the acceptance criteria applicable to this chapter. M&O Licensing can provide copies of the reports for KTI information regarding how the U.S. Department of Energy (DOE) has dispositioned and/or responded to each related issue resolution status report (IRSR).

12.1 PERFORMANCE CONFIRMATION PLAN

Describe in general terms the performance confirmation program. Show that the program has been planned and implemented to satisfy the requirements of 10 CFR 60.140, 141, 142, and 143 for the structures, systems, and components that have been classified as important to isolation. A recommended format is that currently used in the *Performance Confirmation Plan* (CRWMS M&O 1997). Items to be discussed include:

- Objectives from *Performance Confirmation Plan* (CRWMS M&O 1997)

- Overall performance confirmation approach for performance confirmation parameters

- Selection methodology for baseline definition

- Schedule

Discuss the performance confirmation program applicable to the various stages of repository operation from site characterization to permanent closure. Discuss and justify changes to the performance confirmation program as described in the site characterization plan.

12.2 PERFORMANCE CONFIRMATION FOR THE NATURAL BARRIERS

Describe the specific elements of the performance confirmation program pertaining to the natural barriers. Show that the program has been planned and implemented to satisfy the requirements of 10 CFR 60.140, 141, and 142 for the systems, structures, and components that have been classified as important to isolation. For each natural barrier or system of the geologic setting, demonstrate that the performance confirmation program will:

- Not adversely affect the ability of the natural and engineered elements of the geologic repository to meet the postclosure performance objectives
- Monitor and analyze changes from the baseline condition of parameters that could affect the performance of the geologic repository by evaluating measurements and observations against design assumptions and design bases
- Confirm design assumptions and parameters

12.2.1 Geologic System

Describe the performance confirmation program specifically related to the geologic system. Summarize information on the assumed limits of subsurface conditions, geotechnical and design parameters, and original design bases and assumptions relevant to the geologic system that have been presented in detail in Section 3.3 of the License Application. Identify those parameters and natural processes pertaining to the geologic system of the geologic setting that may be changed by site characterization, construction, and operational activities. Highlight parameters that could affect the performance of a geologic repository. 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 from the baseline condition of parameters that could affect the performance of the geologic repository. Provide detailed information on the baseline condition of parameters and natural processes pertaining to the geologic system, including results of site characterization, analytical techniques, experimental data, and any other supporting data in Section 3.3. This section must include the following items:

- Performance confirmation parameters and the bases (justification) for selection
- Performance measures and limits associated with the most significant parameters
- Performance prediction and modeling
- Justification for not including significant parameters that are determined not to be practicable

- General schedules
- General testing plans
- Summary of the baseline condition or parameters and natural processes in tabular format (with reference to Section 3.3 for supporting details)

12.2.2 Hydrologic System

Describe the details of the performance confirmation program specifically related to the hydrologic system. Summarize information on the assumed limits of subsurface conditions, geotechnical and design parameters, and original design bases and assumptions that are relevant to the hydrologic system and that have been presented in detail in Section 3.5 of the License Application. Identify those parameters and natural processes pertaining to the hydrologic system of the geologic setting that may be changed by site characterization, construction, and operational activities. Parameters that could affect the performance of the geologic repository are to be highlighted. Include details of any in situ monitoring, laboratory and field testing, and in situ experiments planned to confirm design assumptions and parameters and to monitor and evaluate changes from the baseline condition of parameters that could affect the performance of the geologic repository. Detailed information on the baseline condition of parameters and natural processes pertaining to the hydrologic system, including results of site characterization, analytical techniques, experimental data, and any other supporting data, are to be described in Section 3.5 of the License Application.

Items that should be included are as follows:

- Performance confirmation parameters and the bases (justification) for their selection
- Performance measures and limits associated with the most significant parameters
- Performance prediction and modeling
- Justification for not including significant parameters that are determined not to be practicable
- General schedules
- General testing plans
- Summary of the baseline condition of parameters in tabular format with reference to Section 3.5 for supporting details

12.2.3 Geochemical System

Describe the details of the performance confirmation program specifically related to the geochemical system. Summarize information on the assumed limits of subsurface conditions, geotechnical and design parameters, and original design bases and assumptions that are relevant

to the geochemical system and that have been presented in Section 3.6 of the License Application. Identify those parameters and natural processes pertaining to the geochemical system of the geologic setting that may be changed by site characterization, construction, and operational activities. Parameters that could affect the performance of the geologic repository are to be highlighted. Include details of any in situ monitoring, laboratory and field testing, and in situ experiments planned to confirm design assumptions and parameters and to monitor and evaluate changes from the baseline condition of parameters that could affect the performance of the geologic repository. Detailed information on the baseline condition of parameters and natural processes pertaining to the geochemical system, including results of site characterization, analytical techniques, experimental data, and any other supporting data are to be described in Section 3.6 of the License Application.

Items that should be included are as follows:

- Performance confirmation parameters and the bases (justification) for selection
- Performance measures and limits associated with the most significant parameters
- Performance prediction and modeling
- Justification for not including significant parameters that are determined not to be practicable
- General schedules
- General testing plans
- Summary of the baseline condition of parameters in tabular format with reference to Section 3.6 for supporting details

12.2.4 Climatological and Meteorological Systems

Describe the performance confirmation program related to the climatological and meteorological systems. Summarize information on climatological, meteorological, and design parameters, as well as original design bases and assumptions that are relevant to the climatological and meteorological systems and that have been presented elsewhere in Section 3.4 of the License Application. Parameters that could affect the performance of the geologic repository are to be highlighted. Include details of any monitoring, testing, or experiments planned to confirm design assumptions and parameters and to monitor and evaluate changes from the baseline condition of parameters that could affect the performance of the geologic repository. Detailed information on the baseline condition of parameters and natural processes pertaining to the climatological and meteorological systems, including results of site characterization, analytic techniques, experimental data, and any other supporting data, are to be described in Section 3.4 of the License Application.

Items that should be included in this section are as follows:

- Performance confirmation parameters and the bases (justification) for selection
- Performance measures and limits associated with the most significant parameters
- Performance prediction and modeling
- Justification for not including significant parameters that are determined not to be practicable
- General schedules
- General testing plans
- Summary of the baseline condition of parameters and natural processes in tabular format with reference to Section 3.4 for supporting details

12.3 PERFORMANCE CONFIRMATION FOR THE SYSTEMS, STRUCTURES, AND COMPONENTS OF THE GEOLOGIC REPOSITORY OPERATIONS AREA

Describe the performance confirmation program to be conducted on natural and engineered systems, structures, and components of the geologic repository operations area. Show that the program has been planned and implemented to satisfy the requirements of 10 CFR 60.140, 141, and 142 for the systems, structures, and components that have been classified as important to isolation.

12.3.1 Surface Facilities

A specific performance confirmation program for surface facilities is not necessary. Describe how the performance confirmation programs for other parts of the program affect design of the surface facilities.

12.3.2 Shafts and Ramps

Provide a description of all performance confirmation testing that will be conducted for the shafts and ramps. Include all stages of planning and implementation of the performance confirmation program for the shafts and ramps, spanning the period from site characterization to just prior to permanent closure. This period should include:

- Collection of baseline data for the penetrated strata and the shaft and ramp engineered components
- Initial performance assessments for the shaft/ramp
- Revising the *Performance Confirmation Plan* (CRWMS M&O 1997) based on data collected during construction and subsequent operations

- Full-scale testing to evaluate the effectiveness of seals, grouts, plugs, and backfill and to evaluate the effectiveness of drainage

- Performance assessments supporting permanent closure

Include discussion of:

- Development of computer codes that will be used for performance confirmation, especially to assess interaction effects of the thermal load on the linings, seals, and backfill

- Facilities where the performance confirmation testing will be conducted

- Component in which the testing is conducted and parameters measured

- Duration of the tests

- Whether testing is conducted using nuclear materials

12.3.3 Underground Facility

Provide a description of performance confirmation testing that will be conducted for the underground facility. Include the stages of planning and implementation of the performance confirmation program for the underground facility, spanning the period from site characterization to just prior to permanent closure. This period should include:

- Collection of baseline data for the host rock and the engineered components

- Initial performance assessments for the underground facility

- Revision of the *Performance Confirmation Plan* (CRWMS M&O 1997) based on data collected during construction and subsequent operations

- Continued performance confirmation to assess thermal and thermomechanical response of the natural and engineered components to construction and emplacement of waste

- Full-scale testing to evaluate the effectiveness of emplacement area seals, grouts, plugs, and backfill

- Performance assessments supporting permanent closure

Include discussion of:

- Development of computer codes that will be used for performance confirmation, especially to assess interaction effects of the thermal load on the linings, seals, and backfill

- Facilities in which the performance confirmation testing will be conducted
- Component in which the testing is conducted and parameters measured
- Duration and sequencing of the tests
- Validation of codes and models
- Whether testing is conducted using nuclear materials

12.4 PERFORMANCE CONFIRMATION FOR THE WASTE PACKAGE AND ENGINEERED BARRIER SYSTEM

Confirmation of performance for the engineered barrier system with regard to anticipated processes and events is required. Describe how the performance confirmation plan will provide this confirmation. Show that the program has been planned and implemented to satisfy the requirements of 10 CFR 60.140, 141, 142, and 143 for the systems, structures, and components that have been classified as important to isolation. Identify those unanticipated processes and events that may lead to failures and that could cause the repository to fail to meet its overall performance requirements.

NRC KTIs

Discuss in general terms how the NRC KTI on container life and source term has been addressed related to Subissue 1. Include a discussion of how NRC sees the issue as discussed in its related IRSR, and provide the DOE perspective on the status of resolution of the issue/subissue/acceptance criteria at the time of preparation of the License Application. Address the following acceptance criterion in a manner that demonstrates resolution of each criterion.

KTI: Container Life and Source Term

Subissue 1: What are the Effects of Corrosion on the Lifetime of the Containers and the Release of Radionuclides to the Near-Field Environment?

This subissue relates to the adequacy of DOE consideration of the effects of corrosion on the lifetime of the containers and the release of radionuclides from the engineered barrier system to the near-field environment.

Acceptance Criteria:

(5) The DOE corrosion testing program is sufficiently complete at the time of the License Application submittal to bound likely effects on performance. In addition, the DOE program identifies specific plans for completion of the testing program to reduce any significant areas of uncertainty as part of the performance confirmation program.

12.4.1 Waste Package Monitoring

Discuss the program established at the geologic repository operations area for monitoring the condition of the waste packages.

12.4.1.1 Waste Form

Describe the performance confirmation program for the waste form as described in the *Performance Confirmation Plan* (CRWMS M&O 1997). (This is not a regulatory requirement; DG-3003 (NRC 1990) states that it is not necessary to have a specific performance confirmation program for the waste form.)

12.4.1.2 In Situ Waste Package Monitoring

Describe the monitoring tests to be performed in situ to observe the performance of the waste packages in their actual repository environment and demonstrate that they conform to design and regulatory requirements. Monitoring tests must address any change in the environment caused by emplacement of the waste packages. The characteristics monitored, in conjunction with external monitoring and laboratory waste package testing, must be adequate to fully describe the components of the waste package for confirming how well the components perform up to the time of repository closure and for validating the methods of extrapolation proposed by DOE to evaluate postclosure performance. Describe how the packages selected are representative of the total waste package population.

12.4.1.3 External Waste Package Monitoring Environment

Describe the methods for monitoring the external environment of the waste packages. Describe how the monitoring data accurately reflects performance under normal repository conditions. Describe how these conditions do not endanger the integrity of the repository or personnel engaged in that testing.

12.4.1.4 Laboratory Waste Package Monitoring

Provide a summary of the waste package monitoring program intended to satisfy 10 CFR 60.143(c), which states: "The waste package monitoring program shall include laboratory experiments which focus on the internal condition of the waste packages. To the extent practical, the environment experienced by the emplaced waste packages within the underground facility during the waste package monitoring program shall be duplicated in the laboratory experiments." Describe the conditions within the waste package to be monitored and the bases for selecting these parameters.

12.4.1.5 Duration of Post-Emplacement Waste Package Monitoring

Provide a schedule for monitoring of the emplaced waste packages. State that the monitoring shall begin immediately after emplacement is initiated and will extend to the end of the retrieval period that is permanent closure. See 10 CFR 60.143(d).

12.4.1.6 Demonstration of Compliance

Describe the program for demonstration of compliance with waste package monitoring requirements. Include environmental parameters to be monitored, waste package condition, types of monitors used, location of monitors, monitor polling frequency, water samples, accuracy of measurement, quality control, calibration, and reliability of maintenance and monitoring systems.

12.4.2 Engineered Barrier and Waste Package Performance Objectives

Discuss the program established for conforming the compliance of the engineering barrier system and waste package with regulatory performance objectives.

12.4.2.1 Failure Mode and Effects Analysis

Discuss the screening of repository scenarios and anticipated and unanticipated processes and events leading to engineered barrier system component failures. For those unanticipated processes and events that may cause failures, the consequences and effects of such failures must be analyzed. The effects from unanticipated engineered barrier system failures are to be presented and shown not to cause failure of the repository to meet performance requirements.

12.4.2.2 Environmental Conditions for Confirmatory Tests and Analyses

Document test conditions and repository environmental conditions for which the engineered barrier system are analyzed for performance confirmation. Relate the conditions to repository scenarios. If the conditions are not the same as those found in the scenarios, provide justification and rationales for using them.

12.4.2.3 Confirmatory Tests and Analysis

Describe the overall confirmatory methodology for the engineered barrier system and the waste package. Document tests and analyses, including any computer codes and methods of extrapolation used. The tests may be any combination of laboratory tests, field tests, and in situ tests, and the analyses may be an interpretation of test results or analogies. In situ tests and other onsite tests must not adversely affect the ability of the natural and engineered elements of the geologic repository to meet the performance objectives.

12.4.2.4 Engineered Barrier System Performance Allocation

Discuss design performance allocated to the different components of the engineered barrier system and how the allocation contributes to satisfaction of the overall performance requirements.

12.4.2.5 Results of Engineered Barrier System Performance Confirmation

Present and discuss results of confirmatory tests and analyses completed at time of submittal of the License Application.

12.5 RADIATION PROTECTION FOR PERFORMANCE CONFIRMATION ACTIVITIES

This section of the License Application should address only radiation protection features specifically related to the performance confirmation program. Radiation protection is described in detail in Chapter 10 of this *Technical Guidance Document for License Application Preparation*. For any systems or portions of systems that are specifically in place to support the performance confirmation program, either provide the information required by 10 CFR 60.21(c)(1)(ii)(E), 60.21(c)(2) and (3), 60.111(a), 60.131(a)-(e), (g), and (i), and 60.136(b), or refer to applicable portions of other chapters of the License Application that provide this information. (It is anticipated that performance confirmation structures, systems, and components will generally be fully integrated into the design of other repository systems and are therefore discussed elsewhere in the License Application. Thus, much of the information required by this section will have been presented elsewhere and should be referenced, rather than repeated. Similarly, radiation control plans and procedures, radiological assessments related to design basis events, and effluent control systems for maintaining releases to the environment as low as reasonably achievable [ALARA] should not be repeated here.)

Demonstrate compliance with the regulatory requirements for radiation protection during performance confirmation. This section of the License Application may include the following:

- An analysis or discussion of analysis of the design and performance of systems, structures, and components, both surface and subsurface, for the performance confirmation program, to identify those that are important to safety. For the purposes of this analysis, it shall be assumed that operations at the geologic repository operations area will be carried out at the maximum capacity and rate of receipt of radioactive waste stated in the application.

- A general description and discussion of the design, both surface and subsurface, as it relates specifically to radiation protection for the performance confirmation program. This description should include:

  - Principal design criteria and relationships to general performance objectives

  - Design bases and the relation of the design bases to the principal design criteria

  - Information relative to materials of construction, including geologic media, general arrangement, and approximate dimensions

  - Codes and standards that DOE proposes to apply to the design and construction of the geologic repository operations area

- System diagrams to support the system descriptions. On the drawings identify and show any radiological design features.

- A description and analysis of the design and performance requirements for systems, structures, and components for the performance confirmation program. This analysis

shall consider the margins of safety under normal operations and those conditions and events that can be reasonably expected to occur before permanent closure.

- An identification and justification for the selection of those variables, conditions, or other items that are determined to be probable subjects of license specifications. Special attention shall be given to those items that may significantly influence the final design for the performance confirmation program.

- A description of the interrelationship of the performance confirmation systems and the low-level radioactive waste management (effluent control and monitoring) systems. Refer to appropriate sections in Chapter 9 for detailed description of the systems for control and monitoring of radioactive effluents to maintain such effluents and exposures, in accordance with the requirements of 10 CFR 20. The information in Chapter 9 should not be reiterated. Cross-reference the information as needed.

- An identification of those systems, structures, and components of the geologic repository, both surface and subsurface for the performance confirmation program, that require research and development to confirm the adequacy of design.

- A description of the elements of the performance confirmation program expected to be available for purposes of radiation protection. Describe the features in sufficient detail to demonstrate that exposures to workers and concentrations of radioactive material in the air will be within the limits specified in 10 CFR 20. Include in the description robotics, portable or specialized shielding, equipment and facilities for handling contaminated materials, special environmental considerations, if appropriate, and other specialized features independent of the systems and subsystems detailed in Chapters 4 and 9. Refer to Chapter 10 for detailed description of the radiation protection program.

10 CFR 20.1101(b) requires that the licensee use, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are ALARA. Regulatory Guide 8.8, *Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable* (NRC 1978), provides information relevant to attaining ALARA goals and objectives. Regulatory Position 2, Facility and Equipment Design Features, contains practices for incorporating ALARA principles into the design of facilities. Make a statement that indicates the extent (if any) that the geologic repository operations area design to support performance confirmation is consistent/compliant with these aspects of regulatory guidance. Describe procedures, engineering controls, and specific design features that apply to performance confirmation system/subsystems that will be incorporated to apply ALARA principles to those systems. Relate specific design features to those discussed in Regulatory Position 2 of Regulatory Guide 8.8 (NRC 1978), as applicable. Identify ALARA features that apply to dose reduction of both the public and the occupational worker.

Cost-benefit analyses are considered by NRC to be an integral part of a viable ALARA program. Lack of cost-benefit analyses related to ALARA design features are considered to be de facto evidence that the ALARA program is not properly implemented. Describe in this section any cost-benefit analyses associated with the design of system/subsystems, if any, for performance

confirmation. Regulatory Guide 1.110, *Cost-Benefit Analysis for Radioactive Waste Systems for Light-Water-Cooled Nuclear Power Reactors* (NRC 1976), provides guidance on acceptable practices for cost-benefit analyses that involve consideration of reduction in dose to the general public. Review this guide. Some of the guidance is potentially useful and applicable. The project will assess the applicability of the guidance in this document, and appropriate instructions for the License Application author will be provided in a future revision of the technical guidance document.

12.6 ANALYSIS OF CHANGES FROM PERFORMANCE CONFIRMATION BASELINE

Describe the plan for monitoring and analyzing changes from the baseline condition of parameters that could affect the performance of a geologic repository. Show that the program has been planned and implemented to satisfy the requirements of 10 CFR 60.141(d). Specifically discuss the process for feedback, analysis of data, and implementation of appropriate action.

The format in the *Performance Confirmation Plan* (CRWMS M&O 1997), Section 3.4, provides a suitable format for this section.

12.7 UNRESOLVED SAFETY QUESTIONS

Taking into account the assessments done in the previous sections, identify all unresolved safety questions, and provide a schedule indicating when they will be resolved. Show that the program has been planned and implemented to satisfy the requirements of 10 CFR 60.21(c)(14).

V. REFERENCES

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

10 CFR 60. Energy: Disposal of High-Level Radioactive Wastes in Geologic Repositories.

CRWMS M&O (Civilian Radioactive Waste Management System Management and Operating Contractor) 1997. *Performance Confirmation Plan.* B00000000-00841-4600-00002. Rev. 0. Las Vegas, Nevada. MOL.19980204.1022.

DOE (U.S. Department of Energy) 1988. *Site Characterization Plan Yucca Mountain Site, Nevada Research and Development Area, Nevada.* DOE/RW-0199. Washington, D.C.: Office of Civilian Radioactive Waste Management. 223856.

DOE 1998. *Quality Assurance Requirements and Description.* DOE/RW-0333P. Rev. 8. Washington, D.C.: U.S. Government Printing Office. MOL.19980601.0022.

NRC (U.S. Nuclear Regulatory Commission) 1976. *Cost-Benefit Analysis for Radioactive Waste Systems for Light-Water-Cooled Nuclear Power Reactors.* Regulatory Guide 1.110. Rev. 0. Washington, D.C.: U.S. Government Printing Office. 2779.

NRC 1978. *Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable.* Regulatory Guide 8.8. Rev. 3. Washington, D.C.: U.S. Government Printing Office. 2887.

NRC 1990. *Format and Content for the License Application for the High-Level Waste Repository.* Regulatory Guide DG-3003. Washington, D.C.: U.S. Government Printing Office. 223107.

NRC 1995. *License Application Review Plan for a Geologic Repository for Spent Nuclear Fuel and High-Level Radioactive Waste.* NUREG-1323. Rev. 1. Washington, D.C.: U.S. Government Printing Office. 213043.

CHAPTER 13. LAND OWNERSHIP AND CONTROL

I. OBJECTIVE

This chapter provides guidance for the author(s) of Chapter 13 of the License Application. The License Application will be submitted to the U.S. Nuclear Regulatory Commission (NRC) for authorization to construct a geologic high-level radioactive waste repository at Yucca Mountain. The author(s) must demonstrate that the U.S. Department of Energy (DOE) has acquired, or will acquire, land and other interests in property needed to meet NRC requirements and to conduct geologic repository operations, including waste isolation.

II. GUIDANCE TOPICS

The following topics will be discussed in Chapter 13 of the License Application:

- 13.0 PURPOSE, SUMMARY, AND CONCLUSIONS
- 13.1 ACQUISITION OF THE CONTROLLED AREA
  - 13.1.1 Identification of the Controlled Area
  - 13.1.2 Existing Legal Interests
  - 13.1.3 Legal Interests to be Obtained
  - 13.1.4 Water Rights Within the Controlled Area
- 13.2 PLANS FOR REGULATING LAND USE OUTSIDE THE CONTROLLED AREA
  - 13.2.1 Identification of Adjacent Areas of Concern
  - 13.2.2 Existing Legal Interests
  - 13.2.3 Legal Interests to be Obtained
- 13.3 PLANS TO CONTROL ACCESS TO THE GEOLOGIC REPOSITORY OPERATIONS AREA

III. REGULATORY REQUIREMENTS

***** IMPORTANT NOTE *****

At the time of development of this document, it appears likely that NRC will issue a new, site-specific regulation for Yucca Mountain in 1999. At the time the new 10 CFR 63 is issued, 10 CFR 60 will be modified to state that it is not applicable to Yucca Mountain. Details about the new regulation are not known. However, it has been determined prudent to develop the working draft License Application in accordance with the best available estimate of the form of the new regulation. The form of the new regulation is not expected to affect the context of the administrative requirements for land ownership and control. Therefore, the text of the technical guidance document that follows this note has not been revised and the working draft License Application authors are expected to follow the guidance provided below. Any questions should be addressed to Management and Operating Contractor (M&O) Licensing.

***** END OF NOTE *****

Sections of the regulations that are to be addressed fully or in part in Chapter 13 of the License Application are reproduced below. These are verbatim extracts from the regulations. Paragraph numbers refer to Title 10 of the U.S. Code of Federal Regulations (CFR). In addition, Appendix A provides a cross-reference between the regulations in 10 CFR 60 and the specific License Application sections that address these regulations. Revisions to regulations are not uncommon. Verify that current versions are used.

10 CFR 60, Disposal of High-Level Radioactive Wastes in Geologic Repositories
Subpart A - General Provisions
60.2 Definitions.

As used in this part—

"Accessible environment" means: (1) The atmosphere, (2) the land surface, (3) surface water, (4) oceans, and (5) the portion of the lithosphere that is outside the controlled area...

"Containment" means the confinement of radioactive waste within a designated boundary...

"Controlled area" means a surface location, to be marked by suitable monuments, extending horizontally no more than 10 kilometers in any direction from the outer boundary of the underground facility, and the underlying subsurface, which area has been committed to use as a geologic repository and from which incompatible activities would be restricted following permanent closure...

"Geologic repository" means a system which is intended to be used for, or may be used for, the disposal of radioactive wastes in excavate geologic media. A geologic repository includes: (1) The geologic repository operations area, and (2) the portion of the geologic setting that provides isolation of the radioactive waste.

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

"Geologic setting" means the geologic, hydrologic, and geochemical systems of the region in which a geologic repository operations area is or may be located...

"Restricted area" means any area access to which is controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials. "Restricted area" shall not include any areas used as residential quarters, although a

separate room or rooms in a residential building may be set apart as a restricted area...

"Site" means the location of the controlled area...

"Underground facility" means the underground structure, including openings and backfill materials, but excluding shafts, boreholes, and their seals.

"Unrestricted area" means any area access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, and any area used for residential quarters.

Subpart B - Licenses
License Applications
60.21 Content of application.

(c) The Safety Analysis Report shall include:

(8) A description of the controls that the applicant will apply to restrict access and to regulate land use at the site and adjacent areas, including a conceptual design of monuments which would be used to identify the controlled area after permanent closure.

60.51 License amendment for permanent closure.

(a) DOE shall submit an application amend the license prior to permanent closure. The submission shall consist of an update of the License Application submitted under § 60.21 and 60.22, including:

(2) A detailed description of the measures to be employed--such as land use controls, construction of monuments, and preservation of records--to regulate or prevent activities that could impair the long-term isolation of emplaced waste within the geologic repository and to assure that relevant information will be preserved for the use of future generations.

60.102 Concepts.

This section provides a functional overview of Subpart E. In the event of any inconsistency with the definitions found in § 60.2, those definitions shall prevail.

(a) *The HLW facility.* NRC exercises licensing and related regulatory authority over those facilities described in section 202 (3) and (4) of the Energy Reorganization Act of 1974. Any of these facilities is designated a *HLW facility.*

(b) *The geologic repository operations area.* (1) This part deals with the exercise of authority with respect to a particular class of HLW facility--namely a *geologic repository operations area.*

(2) *A geologic repository operations area* consists of those surface and subsurface areas that are part of a geologic repository

where radioactive waste handling activities are conducted. The underground structure, including openings and backfill materials, but excluding shafts, boreholes, and their seals, is designated the underground facility.

(c) *Areas related to isolation.* Although the activities subject to regulation under this part are those to be carried out at the geologic repository operations area, the licensing process also considers characteristics of adjacent areas that are defined in other ways. There is to be an area surrounding the underground facility referred to above, which is designated the *controlled area,* within which DOE is to exercise specified controls to prevent adverse human actions following permanent closure. The location of the controlled area is the *site.* The *accessible environment* is the atmosphere, land surface, surface water, oceans, and the portion of the lithosphere that is outside the controlled area. There is an area, designated the *geologic setting*, which includes the geologic, hydrologic, and geochemical systems of the region in which a geologic repository operations area is or may be located. The geologic repository operations area plus the portion of the geologic setting that provides isolation of the radioactive waste make up the *geologic repository.*

60.121 Requirements for ownership and control of interests in land.

(a) *Ownership of land.* (1) Both the geologic repository operations area and the controlled area shall be located in and on lands that are either acquired lands under the jurisdiction and control of DOE, or lands permanently withdrawn and reserved for its use.

(2) These lands shall be held free and clear of all encumbrances, if significant, such as: (i) Rights arising under the general mining laws; (ii) easements for right-of-way; and (iii) all other rights arising under lease, rights of entry, deed, patent, mortgage, appropriation, prescription, or otherwise.

(b) *Additional Controls.* Appropriate controls shall be established outside of the controlled area. DOE shall exercise any jurisdiction and control over surface and subsurface estates necessary to prevent adverse human actions that could significantly reduce the geologic repository's ability to achieve isolation. The rights of DOE may take the form of appropriate possessory interests, servitudes, or withdrawals from location or patent under the general mining laws.

(c) *Water rights.* (1) DOE shall also have obtained such water rights as may be needed to accomplish the purpose of the geologic repository operations area.

(2) Water rights are included in the additional controls to be established under paragraph (b) of this section.

60.136 Preclosure controlled area.

(a) A preclosure controlled area must be established for the geologic repository operations area.

...

(c) The preclosure controlled area may be traversed by a highway, railroad, or waterway, so long as appropriate and effective arrangements are made to control traffic and protect public health and safety.

IV. ACCEPTANCE CRITERIA AND GUIDANCE

General guidance

This section contains general guidance for preparing Chapter 13 of the License Application for construction authorization. In general, the License Application for construction authorization must provide sufficient information for NRC to make a reasonable assurance determination that DOE has identified the interests in property it has obtained, or will obtain, to conduct geologic repository operations, including waste isolation. To support the License Application and the NRC determination of reasonable assurance, this section must provide the following information:

- Methods by which DOE has acquired, or will acquire, the controlled area
- DOE plans for regulating use of the land outside the land withdrawal area
- DOE plans for controlling access to the geologic repository operations area

This chapter must identify and clearly reference regulatory guidelines. It must also clearly identify other appropriate laws, rules, and regulations used to support the License Application. It must clearly document the reasons for use and the ways they have been or will be used to conduct geologic repository operations, including waste isolation and receipt and possession of high-level radioactive waste.

The guidance in this chapter is based on or extracted from NUREG-1323, *License Application Review Plan for a Geologic Repository for Spent Nuclear Fuel and High-Level Radioactive Waste* (NRC 1995), and on *Mined Geologic Disposal System License Application Annotated Outline*, YMP/94-05 (DOE Office of Civilian Radioactive Waste Management, in draft), which is a preliminary, unapproved draft annotated outline that provides history and information that may be useful to the preparer(s) of this chapter. NUREG-1323 (NRC 1995) is largely incomplete and will become inapplicable to Yucca Mountain upon the expected issuance of a new 10 CFR 63 as discussed in Section III above. Information from NUREG-1323 (NRC 1995) considered likely to be applicable under the new regulation is included in the technical guidance document, though the NUREG is not referenced as an authoritative source for such information.

13.0 PURPOSE, SUMMARY, AND CONCLUSIONS

State the purpose of this chapter of the License Application, which is to demonstrate that DOE has acquired, or will acquire, land and other interests in property that are necessary to meet NRC regulations and to conduct geologic repository operations, including waste isolation.

Discuss the organization of the chapter. Section 13.0 is the introduction of the chapter. Section 13.1 identifies the controlled area and how DOE has acquired, or will acquire, it and demonstrates compliance with 10 CFR 60.121(a) and (c). Section 13.2 identifies the areas of concern adjacent to the controlled area, the existing present and future legal interests in these areas, and the legal interests obtained. It demonstrates compliance with 10 CFR 60.121(b) and (c).

Section 13.3 describes how DOE will restrict access to surface areas where radioactive waste will be handled. This section, with its references, demonstrates that DOE can control access, as necessary, to protect individuals from radiation and radioactive materials. Also identified are those lands constituting the geologic repository and adjacent areas for which DOE has rights-of-access and that are available accordingly to NRC to carry out its inspection activities.

13.1 ACQUISITION OF THE CONTROLLED AREA

This section addresses how DOE has acquired, or will acquire, ownership of the controlled area. It demonstrates compliance with 10 CFR 60.121(a) by showing that the controlled area and the geologic repository operations area are, or will be, free and clear of all significant encumbrances under the jurisdiction of DOE. It also demonstrates compliance with 10 CFR 60.121(c) by showing the sufficiency of water rights with the controlled area obtained by DOE. The following subsections identify and discuss the controlled area, the existing legal interests, legal interests to be obtained, and water rights.

Discuss why the ownership and control status of the land area of interest is important to DOE. Describe land management plans and agreements, and discuss the public domain status of the Bureau of Land Management (BLM) controlled lands. Discuss the laws and regulations that may impact land ownership and control, or the rights to access such lands.

Review Section 9.1 of the *Mined Geologic Disposal System License Application Annotated Outline,* YMP/94-05 (DOE Office of Civilian Radioactive Waste Management, in draft), for suggested wording for this section of the License Application.

13.1.1 Identification of the Controlled Area

Define and describe the preclosure controlled area to comply with 10 CFR 60.136(a), and the postclosure controlled area, as discussed in 10 CFR 60.121. Define the limits of these areas in relation to the accessible environment and refer to appropriate regulations. Discuss the controlled area boundary and identify the location from which the cumulative releases of radionuclides after closure will be calculated to determine compliance with appropriate regulations. Define the restricted area and refer to appropriate regulations.

According to 10 CFR 60.21(c)(8), the postclosure controlled area will be permanently marked with monuments. Discuss how the boundary will be marked and refer to appropriate regulations. Refer to Section 11.7 of the License Application for further discussion of permanent site markers.

Provide the complete, detailed, legal description of the controlled area that conforms to other accepted methods of land description. Provide complete BLM master title plats for all sections contained within the controlled area to ensure they identify existing or proposed title control and existing encumbrances.

Provide an overview of the relevant features within and outside of the controlled area consisting of the natural systems of the geologic setting, the physical facilities of the geologic repository operations area, and the engineered barrier systems. Limit the discussion to those features and facilities that are appropriate to demonstrate the relationship to the size and boundaries of the area for which ownership and control of interests need to be established. In addition, include maps displaying relevant features within and outside the controlled area. Show on these maps important features of the natural systems of the geologic setting, the physical facilities of the geologic repository operations area, and the engineered barrier systems, with appropriate notations. Provide cross-references to chapters that discuss these features in greater depth. Demonstrate that the size and boundaries of the geologic repository operations area and the controlled area and accessible environment are appropriately tied to specific design or natural features and are consistent with the technical justification. Site characteristics discussed in Chapter 3 of the License Application should be incorporated by reference and summarized as necessary in this chapter. The technical justification for the geologic repository operations area design contained in Chapter 4 of the License Application should be only incorporated by reference in this chapter. Detailed analysis of the engineered barrier systems contained in Chapter 6 of the License Application should be incorporated by reference in the description presented in this section. The author should coordinate with the authors of Chapters 3, 4, and 6 to ensure consistency is maintained among the chapters and that unnecessary repetition is avoided.

Identify the limits of the geologic repository operations area and its relationship to the limits of the controlled area on master title plats, maps, and diagrams to show the relevant and necessary controls to achieve waste isolation. If DOE proposes that controls for the geologic repository operations area be different from those for the remainder of the controlled area, describe the extent of the difference and provide appropriate rationale.

Describe limits of the underground facility and maximum horizontal distance from the underground facility to the boundary of the controlled area. Provide a graphic representation, as appropriate.

13.1.2 Existing Legal Interests

Identify any encumbrances currently located on the lands to be included in the controlled area. 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/or prescription

Because the regulations specify "encumbrances, if significant" and "appropriate controls," discuss not only the existence of rights, but also the extent to which they may impact operations and isolation. Cross-reference Section 13.1.3 for plans for extinguishing or compensating existing rights or interests. Refer to sections of the License Application relating to evidence of mining and drilling for natural resources to demonstrate that past mining and drilling activities do not pose a threat to DOE land ownership or to waste isolation.

Provide or refer to the legal documentation of ownership, including sufficient indexes of ownership and control, to satisfy a purchaser-of-record. This documentation shall include, but not be limited to, recorded title search showing any and all interests in land, and BLM master title plat, which indicates all recorded interests and claims.

13.1.3 Legal Interests to be Obtained

Identify the legal interests to be obtained and describe the interests in real property that will be obtained by DOE to demonstrate compliance with the regulatory requirements of 10 CFR 60. Describe how existing legal interests in the controlled area will be extinguished. Demonstrate that significant legal encumbrances do not or will not exist as required by 10 CFR 60.121(a).

Describe how DOE has either acquired or has permanently withdrawn the land that would comprise the geologic repository operations area and controlled area. It is anticipated that necessary land will be withdrawn by an act of Congress. Describe the process by which this has occurred, or will occur.

Include or refer to a copy of the withdrawal legislation. Ensure that the legal descriptions of the land area contained in the statute and the description in the application agree. Since the land area of the proposed repository site would be totally in federal ownership, the statute will constitute complete ownership documentation, subject to subordinate interests.

If statutory withdrawal has not been enacted, show that DOE has taken all steps within its control to establish effective and permanent jurisdiction and control, and legislative or other transfer activities underway will be completed prior to the completion of the NRC review and decision on the application.

Describe the means, such as title search and BLM records search, that will identify any existing or future encumbrances or other surface or subsurface interests of record. Show that the method chosen will be adequate and complete to ensure no competing interests remain and that adverse human intrusion would not be facilitated.

13.1.4 Water Rights Within the Controlled Area

In this section demonstrate compliance with 10 CFR 60.121(c) with regard to water rights within the controlled area. Discuss the water rights determined necessary for operations and to prevent adverse impact on isolation. Address both the potential water needs of the geologic repository operations area, as described in Section 4.1 of the License Application, and the potential adverse impacts on isolation which may result from water usage on the lands within the controlled area. Also describe potential adverse water usage impacts identified outside the controlled area, as described in Section 3.5 of the License Application. Include discussion of existing and projected water uses authorized under existing law and justification for the quantity projections from Chapter 4 of the License Application, as well as potential consequences of estimated water usage both within and outside the controlled area. Summarize water needs for personnel, equipment, fire protection, and other uses deemed necessary.

Describe needed permits and transfers with the focus on those identified as well as those not already obtained. Provide appropriate schedules and show they are adequate to meet project schedules for geologic repository operations area development and waste emplacement. Identify water rights for both the time period of operations and the period after permanent closure. Clearly identify all needed permits and transfers. For those not already obtained, provide a schedule for anticipated approval. The information on water rights and water use in this chapter should be cross-referenced with information submitted under Section 3.5 of the License Application.

13.2 PLANS FOR REGULATING LAND USE OUTSIDE THE CONTROLLED AREA

In this section, demonstrate compliance with 10 CFR 60.121(b). Explain, to the extent practical, such measures necessary that demonstrate how DOE plans to regulate land use outside the controlled area. If the controlled area is sufficiently large to make regulation of surrounding areas unnecessary, so state and provide a rationale. This information will assist NRC in judging whether the repository complies with 10 CFR 60.121(b), which states:

> Appropriate controls shall be established outside of the controlled area. DOE shall exercise any jurisdiction and control over surface and subsurface estates necessary to prevent adverse human actions that could significantly reduce the geologic repository's ability to achieve isolation. The rights of DOE may take the form of appropriate possessory interests, servitude, or withdrawals from location or patent under the general mining laws.

Since NRC does not explain "appropriate" land controls, the term may conservatively be assumed to mean the same type of controls that NRC requires for the land within the controlled area. In the following sections, demonstrate the adequacy of land controls for this area.

13.2.1 Identification of Adjacent Areas of Concern

Identify by means of description and a map all areas adjacent to the controlled area. Describe the areas within this set of areas that are deemed to be of concern, and explain the basis for the

concern. Appropriately tie the size and boundaries of the areas of concern outside the controlled area to specific design or natural features. The description of the areas should also be consistent with the technical justification to ensure the repository's ability to achieve isolation and to prevent or reduce the risk of human activity, which may adversely impact isolation. Describe human actions outside the controlled area considered to have the potential to affect the geologic repository and explain how this potential affects designation of areas of concern.

Provide the legal description of areas of concern outside the controlled area.

13.2.2 Existing Legal Interests

Explain the existing legal interests in the areas of concern outside the controlled area. Show that the legal documentation of ownership and/or control includes sufficient indexes of ownership and control of the area outside the controlled area as to satisfy a purchaser-of-record including, but not limited to, a recorded title search showing any and all interests in land, and the BLM master title plat, which indicates all recorded interests and claims.

Describe existing interests of the following types:

- Rights arising under the general mining laws
- Easements for rights-of-way
- Rights arising under lease, rights of entry, deed, patent, mortgage, appropriation, and/or prescription (excluding rights arising under the general mining laws and easements for rights-of-way)

Divide this discussion into:

- Present surface interests
- Future surface interests
- Present subsurface interests
- Future subsurface interests

For each right, describe the nature of activities which may permissibly occur under the rights.

Describe existing and future water rights in the areas of concern. Include the nature of the right, the intended use of the water, and provide quantitative information on the water use expected or allowed. Discuss the concern the water rights pose for waste isolation and repository operation. Also discuss whether obtaining the water right is necessary to support repository operations. Provide quantitative information by water category regarding water rights within the adjacent areas of concern outside the land withdrawal area.

Describe the extent to ,which, in the absence of unanticipated processes and events, DOE jurisdiction and control of the legal interests would serve to prevent adverse human actions that could significantly reduce the geologic repository's capability for isolation. Reference Section 3.5 of the License Application, which assesses the impacts of human activities on

groundwater, and Section 3.3.10, which assesses the impacts of mining and drilling for natural resources.

List relevant existing legal records pertaining to the legal interests defined in this section, and identify locations in a table format.

13.2.3 Legal Interests to be Obtained

Describe the legal interests obtained or planned to be obtained in the adjacent areas of concern to show compliance with 10 CFR 60.131(b) and with 10 CFR 60.131(c) for water rights outside the controlled area.

List the statutes applicable to the legal interests to be obtained within the adjacent areas of concern outside the controlled area. Reference judicial decisions applicable to the legal interests to be obtained.

Identify encumbrances that DOE proposes not to acquire within the adjacent areas of concern outside the controlled area. Explain why these encumbrances are not significant to the achievement of the purposes of 10 CFR 60.

Identify encumbrances, including water rights, DOE plans to acquire. List relevant legal records to be obtained pertaining to the legal interests to be obtained and locations. Describe the DOE plans for obtaining these rights. Discuss what type of control is to be exercised in each case.

Identify the means, such as title search and BLM records search, that will identify any existing or future encumbrances or other surface or subsurface interests of record in the land area outside the controlled area. Show that the method chosen is adequate and complete to ensure no competing interests remain, or that adverse human intrusion would not be facilitated.

If a statutory withdrawal has been enacted, include or refer to a copy of the legislation and ensure that the legal descriptions of the land area contained in the statute and the description in the License Application agree. Since the land area of the proposed repository site would be totally in federal ownership, the statute will constitute complete ownership documentation, subject to subordinate interests.

If statutory withdrawal has not been enacted, show that DOE has taken all steps within its control to establish effective and permanent jurisdiction and control, and legislative or other transfer activities underway will be completed prior to the completion of the NRC review and decision on the License Application.

13.3 PLANS TO CONTROL ACCESS TO THE GEOLOGIC REPOSITORY OPERATIONS AREA

Describe in this section the precautions that DOE has taken or will take to restrict access to surface areas where radioactive waste is handled, the portion of the geologic repository operations area that is aboveground during the period prior to closure. This area is defined as the "restricted area" in 10 CFR 60.2.

Describe the restricted area in terms of topography and location. Distinguish the location and purpose of the restricted area from the following other areas:

- Unrestricted area
- Underground facility (subsurface boundary)
- Disturbed zone (subsurface boundary)
- Preclosure controlled area (where DOE would calculate doses of a design-basis accident)
- Postclosure controlled area (where DOE would calculate cumulative postclosure releases)
- Land withdrawal area (land which Congress will have given DOE)

Describe the access control measures established for the restricted area and adjacent land that DOE has obtained to show compliance with 10 CFR 60.21(c)(8). Cross-reference Section 1.5 of the License Application for information on the security measures that will limit access to the controlled area.

Describe specific measures to protect individuals in the unrestricted area.

10 CFR 60.136(c) allows the preclosure controlled area to be traversed by a highway, railroad, or waterway as long as appropriate and effective arrangements are made to control traffic and protect public health and safety. Describe the existence of any such transportation routes and explain arrangements made to demonstrate compliance with this regulation. Use map(s) to illustrate the relationship between these routes and the geologic repository operations area.

Discuss planned land use controls other than those already presented in Sections 13.1 and 13.2 of the License Application to regulate or prevent activities that could impair long-term isolation of emplaced waste to preliminarily demonstrate compliance with 10 CFR 60.51(a)(2), a requirement for the License Application amendment for permanent closure.

For the postclosure period, the controlled area will be identified by markers or monuments that would "identify the location of and control access to, *those areas necessary to achieve geologic repository operations*" (emphasis added, NRC 1995).[1] A conceptual design of these markers is to be presented in Section 11.7 of the License Application; briefly summarize use of such markers in this section. Provide a cross-reference to Section 11.7. The final design details of these markers will be found in the subsequent application to decommission and permanently close the geologic repository.

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[1] The italicized passage is interpreted to mean "those areas that were once necessary to achieve geologic repository operations." Markers and monuments would not be erected while the repository is still operating but after it has been permanently closed.

V. REFERENCES

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

NRC (U.S. Nuclear Regulatory Commission) 1995. *License Application Review Plan for a Geologic Repository for Spent Nuclear Fuel and High-Level Radioactive Waste.* NUREG-1323. Rev. 1. Washington, D.C.: U.S. Government Printing Office. 213043.

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CHAPTER 14. QUALITY ASSURANCE

I. OBJECTIVE

This chapter provides guidance for the author(s) of Chapter 14 of the License Application. The License Application will be submitted to the U.S. Nuclear Regulatory Commission (NRC) for authorization to construct a geologic high-level radioactive waste repository at Yucca Mountain. Therefore, at the time the update is submitted for the license to receive and possess radioactive waste, additional information should be minimal in scope.

The author(s) must demonstrate, in a manner that complies with requirements of 10 CFR 60, Subpart B, License/License Applications, that the QA program established and executed for various activities associated with the geologic repository meet the requirements of 10 CFR 60, Subpart G, Quality Assurance, and that the completed and ongoing quality affecting activities at the time of License Application were determined (e.g., by audits and surveillances) to comply with site characterization QA program requirements.

Chapter 14 author(s) must coordinate with other chapter authors to ensure that applicable QA program implementation elements are sufficiently discussed in other parts of the License Application to demonstrate compliance with QA-related guidance and acceptance criteria that are not addressed in detail in this chapter.

The Office of Civilian Radioactive Waste Management (OCRWM) QA program is embodied in the *Quality Assurance Requirements and Description* (DOE 1998). Each revision has been reviewed and accepted by NRC. Crosswalks between DOE 1998 and all applicable U.S. Code of Federal Regulations (CFRs), NUREGs, and regulatory guides have been prepared and submitted to NRC at earlier dates to obtain acceptance. NRC acceptance signifies that the documented QA program complies with the requirements of 10 CFR 60, Subpart G. Accordingly, the U.S. Department of Energy (DOE) has determined that the author(s) need not describe the QA program in detail in this chapter. This practice is consistent with some commercial nuclear power plant Safety Analysis Reports.

II. GUIDANCE TOPICS

The following topics are to be discussed in Chapter 14 of the License Application:

14.0 PURPOSE, SUMMARY, AND CONCLUSIONS
14.1 DESCRIPTION OF QA PROGRAM
- 14.1.1 Compliance with Subpart G of 10 CFR 60
- 14.1.2 Classification of Items (Structures, Systems, and Components)
- 14.1.3 Classification of Activities
- 14.1.4 Quality Assurance Issues

14.2 IMPLEMENTATION OF THE SITE CHARACTERIZATION QA PROGRAM

III. REGULATORY REQUIREMENTS

***** IMPORTANT NOTE *****

At the time of development of this document, it appears likely that NRC will issue a new, site-specific regulation for Yucca Mountain in 1999. When the new 10 CFR 63 is issued, 10 CFR 60 will be modified to state that it is not applicable to Yucca Mountain. Details about the new regulation are not known. However, it has been determined prudent to develop the working draft License Application in accordance with the best available estimate of the form of the new regulation. To comply with management direction, to avoid confusing references, and to allow production of the technical guidance document in time to support training for the working draft License Application, the text of the technical guidance document that follows this note has not been revised to reflect the postulated form of the new regulation. Requirements associated with 10 CFR 60.131 have been deleted, as that section is assumed not to be carried over to the new regulation.

***** END OF NOTE *****

Sections of the regulations that are to be addressed fully or in part in Chapter 14 of the License Application are reproduced below. These are verbatim extracts from the regulations. Paragraph numbers refer to Title 10 of the CFR. In addition, Appendix A provides a cross-reference between the 10 CFR 60 regulations and the specific License Application sections that address these regulations. Revisions to regulations are not uncommon. Verify that current versions are used.

Continual NRC acceptance of the OCRWM QA program (DOE 1998) has established that regulatory, commitment, and guidance documents applicable to this chapter have been met, with the exception of 10 CFR 60, Subpart B. Therefore, the author(s) need only address the regulatory requirements pertaining to License Application preparation.

10 CFR Part 60, Disposal of High-Level Radioactive Wastes in Geologic Repositories
Subpart B - Licenses
License Applications
60.21 Content of application.

(c) The Safety Analysis Report shall include:

(4) A description of the quality assurance program to be applied to the structures, systems, and components important to safety and to the engineered and natural barriers important to waste isolation.

60.31 Construction authorization.

(a) Safety. That there is reasonable assurance that the types and amounts of radioactive materials described in the application can be received, possessed, and disposed of in a geologic repository operations area of the design proposed without unreasonable risk to the health and safety of the public. In arriving at this determination, the Commission shall consider whether:

(3) The DOE's quality assurance program complies with the requirements of Subpart G of this part.

Subpart G – Quality Assurance
60.150 Scope.

As used in this part, *quality assurance* comprises all those planned and systematic actions necessary to provide adequate confidence that the geologic repository and its subsystems or components will perform satisfactorily in service. Quality assurance includes quality control, which comprises those quality assurance actions related to the physical characteristics of a material, structure, component, or system which provide a means to control the quality of the material, structure, component, or system to predetermined requirements.

60.151 Applicability.

The quality assurance program applies to all systems, structures and components important to safety, to design and characterization of barriers important to waste isolation and to activities related thereto. These activities include: site characterization, facility and equipment construction, facility operation, performance confirmation, permanent closure, and decontamination and dismantling of surface facilities.

60.152 Implementation.

The DOE shall implement a quality assurance program based on the criteria of Appendix B of 10 CFR Part 50 as applicable, and appropriately supplemented by additional criteria as required by §60.151.

IV. ACCEPTANCE CRITERIA AND GUIDANCE

This section identifies the acceptance criteria and guidance to be used in developing an acceptable and sufficient QA chapter for the License Application. The guidance in this chapter is generally based on or extracted from the following documents:

- Regulatory Guide DG-3003, *Format and Content for the License Application for the High-Level Waste Repository*, Chapter 10, Quality Assurance (NRC 1990)
- NUREG-1323, *License Application Review Plan for a Geologic Repository for Spent Nuclear Fuel and High-Level Radioactive Waste*, Chapter 10, Quality Assurance (NRC 1995)

NUREG-1323 (NRC 1995) is largely incomplete and will become inapplicable to Yucca Mountain upon the expected issuance of a new 10 CFR 63 as discussed in Section III above. Information from NUREG-1323 (NRC 1995) considered likely to be applicable under the new regulation is included in the technical guidance document, though the NUREG is not referenced as an authoritative source for such information.

As is the case for NUREG-1323 (NRC 1995), DG-3003 (NRC 1990) will become inapplicable to Yucca Mountain upon the issuance of a new 10 CFR 63. Information from DG-3003 (NRC 1990) considered likely to be applicable under the new regulation is included in the technical guidance document, though DG-3003 is not referenced as an authoritative source for such information.

In general, sufficient information must be provided, either in this chapter or supplemented by references to other parts of the License Application, such that NRC will be able to perform an acceptance review and a compliance or safety review with satisfactory results.

Acceptance Review

The following acceptance criteria are extracted from Chapter 10 of NUREG-1323 (NRC 1995):

- To ensure that this section of the License Application is acceptable for docketing, provide the following information:
  - Describe the QA program established and executed for various activities associated with the geologic repository to meet the requirements of Subpart G to 10 CFR 60.

NOTE: The DOE Office of Quality Assurance regards NRC acceptance of the OCRWM QA program as evidence of compliance with Subpart G to 10 CFR 60. Therefore, the need for a detailed QA program description is precluded.

- Identify the structures, systems, and components important to safety for which the QA program applies, and describe, or reference from another License Application section, the analyses used for this identification. State that the above items and descriptions are incorporated into the 10 CFR 60, Subpart G, QA program.

- Reference the quality activities list (Q-List) (DOE/YMP 1998), which lists major site characterization and design and construction activities under the respective QA program, where provided in the License Application.

- Describe or reference the QA program activities that will ensure compliance with 10 CFR 60.131(a) activities (e.g., protection of worker health) and to items related to those activities.

- Describe assessments of activities during site characterization and their compliance with QA program requirements.

- Provide a listing of topics to be addressed in this chapter, including:

  1. Description of the QA program (for site characterization and for design and construction)

  2. Implementation of the QA program for site characterization

- Describe the QA program applied to activities affecting quality during site characterization of the geologic repository, and describe the QA program that will be applied to the structures, systems, and components important to safety and to the engineered and natural barriers important to waste isolation during the design and construction of the geologic repository.

- Provide a reference to, or a general listing by activity of, existing data that has not been gathered under a 10 CFR 60, Subpart G, QA program and requires qualification for use in licensing.

- Describe how the completed and ongoing quality affecting activities at the time of the License Application submittal were determined (e.g., by audits and surveillances) to comply with the QA program requirements for site characterization.

- The *Q-List* (DOE/YMP 1998) identifies structures, systems, and components important to safety, barriers important to waste isolation and related activities, and the radiological protection aspects of worker health and safety. The analyses used to develop the *Q-List* (DOE/YMP 1998) is consistent with NUREG-1318 (NRC 1988c).

Compliance Review (Safety Review)

The following acceptance criteria represent NRC expectations in the compliance (safety) review process.

In conducting the safety review, NRC will determine whether the information presented in the License Application together with its references provides acceptable demonstration of compliance with applicable regulatory requirements. Specifically, DOE will need to:

- Describe the QA programs applied to site characterization, and to be applied to design and construction

- Provide information illustrating how completed and ongoing quality affecting activities at the time of the License Application submittal were determined to comply with site characterization QA program requirements in sufficient detail to enable NRC to determine whether and how all the applicable requirements of 10 CFR 50, Appendix B, were satisfied

To support the safety review of the implementation of the site characterization program, discuss the following or reference the applicable License Application chapter addressing the following:

- All data and data analyses contributing to the License Application have been developed or qualified under an acceptable QA program. State that data used in the period of site characterization to support the License Application that were not collected under a QA program meeting 10 CFR 60, Subpart G, have been qualified in a manner consistent with the guidance of NUREG-1298 (NRC 1988b).

- Scientific and engineering codes have been adequately documented in terms of verification and model validation. State that scientific and engineering codes used during the period of site characterization have been documented, verified, and validated consistent with the guidance of NUREG-0856 (NRC 1983).

- Peer reviews have been conducted in accordance with QA program requirements. State that peer reviews, where applied, have been conducted consistent with the guidance of NUREG-1297 (NRC 1988a).

The compliance determinations undertaken by NRC will consider whether the acceptance criteria for each of the following compliance reviews have been met. The acceptance criteria have been incorporated into the preceding bullets where applicable:

- Safety review of 10 CFR 60.21(c)(4)
- Safety review of 10 CFR 60.150
- Safety review of 10 CFR 60.151
- Safety review of 10 CFR 60.152

14.0 PURPOSE, SUMMARY, AND CONCLUSIONS

Provide the purpose of Chapter 14 of the License Application, which is to describe the QA program established and executed for various activities associated with the geologic repository to meet the requirements of 10 CFR 60, Subpart G. State that the OCRWM QA program, as embodied in the *Quality Assurance Requirements and Description* (DOE 1998), has been reviewed and accepted by NRC. State that, accordingly, and per direction from the OCRWM Office of Quality Assurance, this chapter will not include a detailed description of the OCRWM QA program. Instead, this chapter will focus on complying with content requirements of regulatory guidance, and confirming that the completed and ongoing quality affecting activities at the time of the License Application submittal were determined to comply with the QA program requirements for site characterization.

Provide a summary of information contained in the chapter and a discussion of the chapter organization. Provide a concluding statement of the overall effect of the information contained in Chapter 14 as it relates to demonstrating compliance with the accepted QA program. State, for example, "the information provided in this chapter will confirm that a QA program meeting the requirements of 10 CFR 60, Subpart G, has been applied to activities affecting quality during site characterization of the proposed geologic repository."

14.1 DESCRIPTION OF QUALITY ASSURANCE PROGRAM

State that the OCRWM QA program is described in the OCRWM *Quality Assurance Requirements and* Description (DOE 1998) document. This document provides policy and requirements for development and implementation of consistent QA procedures at every level of the OCRWM program. Each participant organization is required to develop implementing documents that translate applicable requirements into work processes, and must prepare a matrix identifying where requirements are contained in implementing documents.

State that the QA programs for site characterization and for design and construction are the same, and that the singular program is embodied in the *Quality Assurance Requirements and Description* (DOE 1998). It describes the QA program that was applied to activities affecting quality during site characterization of the geologic repository, and the QA program that will be applied to the structures, systems, and components important to safety and to the engineered and natural barriers important to waste isolation during the design and construction phase of the geologic repository.

14.1.1 Compliance with Subpart G of 10 CFR 60

State that the OCRWM QA program is defined by the *Quality Assurance Requirements and Description* (DOE 1998) document.

State that NRC has reviewed each revision of DOE 1998, and has continually accepted the document based on its compliance with the requirements of 10 CFR 60, Subpart G; 10 CFR 50, Appendix B; 10 CFR 71, Subpart H; and 10 CFR 72 , Subpart G.

Provide a reference to, or general listing by activity of, the data that has not been gathered under a 10 CFR 60, Subpart G, QA program, and therefore requires qualification for licensing.

14.1.2 Classification of Items (Structures, Systems, and Components)

State that the OCRWM QA program identifies the categories for classifying structures, systems, and components (items) for the application of QA controls and development of Q-Lists. The categories of classification are described in QAP-2-3 (use current revision at date of the License Application submittal), *Classification of Permanent Items* (CRWMS M&O 1997). These categories have been applied as appropriate to ensure adequate compliance with QA program requirements.

State that the analyses used to develop the *Q-List* (DOE/YMP 1998) are consistent with the guidance of NUREG-1318 (NRC 1988c).

Verify that the authors of Chapters 2, 4, 5, 6, and 9 of the License Application have included a reference as to which analyses in QAP-2-3 (CRWMS M&O 1997) were used for the classification of various structures, systems, and components important to radiological safety and waste isolation.

14.1.3 Classification of Activities

State that in addition to the "items" discussed previously, "activities" related to these items and activities during site characterization were classified as subject to the QA program (DOE 1998, Section 2.2.3), as summarized below.

- Activities that apply to site characterization data and samples, including activities related to sample collection and the collection and analysis of data to support performance confirmation or performance assessments

- Activities related to the items on a Q-List, such as design, procurement, construction, fabrication, production, handling, packaging, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair, modification, and decontamination

- Activities that provide data used to assess the potential dispersion of radioactive materials from the licensed facility

- Activities related to the high-level radioactive waste form development through qualification, production, and acceptance

- Activities associated with characterization, conditioning, and acceptance of DOE spent nuclear fuel

14.1.4 Quality Assurance Issues

Before the receipt of the License Application, NRC will have conducted pre-licensing reviews of the DOE QA program, including technical reviews and QA reviews and audits. NRC will have

documented concerns resulting from these pre-License Application reviews as open items. Some of these open items, referred to as objections to License Application submittal, may be critical to the NRC License Application review because any lack of acceptable DOE resolution would prevent NRC from conducting a meaningful review. Therefore, as part of its acceptance review for docketing, NRC will evaluate the significance of any unresolved objection to License Application submittal with respect to the effective conduct of licensing activities.

Since data qualification has been a longstanding NRC issue, provide guidance to the author of Chapter 2 of the License Application to include a statement to the effect that "any data requiring qualification was done in accordance with the OCRWM QA program." Ensure that any data included in the License Application which was not gathered under a QA program meeting 10 CFR 60, Subpart G, requirements or not qualified consistent with NUREG-1298 (NRC 1988b) guidance are clearly identified.

Provide a description and resolution of any other QA issues, including those that may have emerged between DOE and NRC in the period just prior to submittal of the License Application.

Verify that the authors of applicable chapters of the License Application have addressed:

- The documentation of scientific and engineering codes, in terms of verification and model validation, consistent with guidance from NUREG-0856 (NRC 1983)
- The conduct of peer reviews consistent with NUREG-1297 (NRC 1988a)
- The conduct of expert elicitations consistent with the guidance of NUREG-1563 (NRC 1996)

14.2 IMPLEMENTATION OF THE SITE CHARACTERIZATION QUALITY ASSURANCE PROGRAM

Describe how the completed and ongoing activities subject to the requirements of the *Quality Assurance Requirements and Description* (DOE 1998) at the time of License Application are determined to comply with the QA program requirements for site characterization. Describe the audits and surveillance programs and how they were applied to ensure effective implementation of the QA program.

Describe assessments of activities during site characterization and their compliance with QA program requirements.

V. REFERENCES

10 CFR 50 (U.S. Code of Federal Regulations). Energy: Domestic Licensing of Production and utilization Facilities.

10 CFR 60. Energy: Disposal of High-Level Radioactive Wastes in Geologic Repositories.

10 CFR 71. Energy: Packaging and Transportation of Radioactive Materials.

10 CFR 72. Energy: Licensing Requirements for the Independent Storage of Spent Nuclear Field and High-Level Radioactive Waste.

CRWMS M&O (Civilian Radioactive Waste Management System Management and Operating Contractor) 1997. *Classification of Permanent Items.* QAP-2-3. Rev. 8. Las Vegas, Nevada. MOL.19980511.0132.

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

DOE/YMP (U.S. Department of Energy/Yucca Mountain Site Characterization Project) 1998. *Q-List.* YMP/90-55Q. Rev. 5. Washington, D.C.: U.S. Department of Energy/Office of Civilian Radioactive Waste Management. 231119, MOL.19980513.0132.

NRC (U.S. Nuclear Regulatory Commission) 1988a. *Peer Review for High-Level Nuclear Waste Repositories.* NUREG-1297. Washington, D.C.: U.S. Government Printing Office. 200651.

NRC 1988b. *Qualification of Existing Data for High-Level Nuclear Waste Repositories.* NUREG-1298. Washington, D.C.: U.S. Government Printing Office. Document on order.

NRC 1988c. *Technical Position on Items and Activities in the High-Level Waste Geologic Repository Program Subject to Quality Assurance Requirements.* NUREG-1318. Washington, D.C.: U.S. Government Printing Office. 200650.

NRC 1990. *Format and Content for the License Application for the High-Level Waste Repository.* Regulatory Guide DG-3003. Washington, D.C.: U.S. Government Printing Office. 223107.

NRC 1983. *Final Technical Position on Documentation of Computer Codes for High-Level Waste Management.* NUREG-0856. Washington, D.C.: U.S. Government Printing Office. 205063.

NRC 1995. *License Application Review Plan for a Geologic Repository for Spent Nuclear Fuel and High-Level Radioactive Waste.* NUREG-1323. Rev. 1. Washington, D.C.: U.S. Government Printing Office. 213043.

NRC 1996. *Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program.* NUREG-1563. Washington, D.C.: U.S. Government Printing Office. 226832.