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Passive Plant Designs
Chapters 2–13

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ABSTRACT

The Electric Power Research Institute (EPRI) is preparing a compendium of technical requirements, referred to as the "Advanced Light Water Reactor [ALWR] Utility Requirements Document," that is acceptable to the design of an ALWR power plant. When completed, this document is intended to be a comprehensive statement of utility requirements for the design, construction, and performance of an ALWR power plant for the 1990s and beyond.

The Requirements Document consists of three volumes. Volume I, "ALWR Policy and Summary of Top-Tier Requirements," is a management-level synopsis of the Requirements Document, including the design objectives and philosophy, the overall physical configuration and features of a future nuclear plant design, and the steps necessary to take the proposed ALWR design criteria beyond the conceptual design state to a completed, functioning power plant. Volume II consists of 13 chapters and contains utility design requirements for an evolutionary nuclear power plant [approximately 1350 megawatts-electric (MWe)]. Volume III contains utility design requirements for nuclear plants for which passive features will be used in their designs (approximately 600 MWe).

In April 1992, the staff of the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, issued Volume 1 and Volume 2 (Parts 1 and 2) of its safety evaluation report (SER) to document the results of its review of Volumes I and 2 of the Requirements Document. Volume I, "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document - Program Summary," provided a discussion of the overall purpose and scope of the Requirements Document, the background of the staff's review, the review approach used by the staff, and a summary of the policy and technical issues raised by the staff during its review. Volume 2, "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document - Evolutionary Plant Designs," gave the results of the staff's review of the 13 chapters of the Requirements Document for evolutionary plant designs.

The staff has prepared Volume 3, "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Requirements Document - Passive Plant Designs," gives the results of the staff's review of the 13 chapters of the Requirements Document for passive plant designs. A preliminary draft of Volume 3 was forwarded to the Commission and the Advisory Committee on Reactor Safeguards (ACRS) in August 1993.

In staff requirements memoranda (SRM), the Commission instructed the staff to provide an analysis detailing where the staff proposes departure from current regulations or where the staff is substantially supplementing or revising interpretive guidance applied to currently licensed LWRs. The staff considers these to be policy issues. Appendix B to Chapter 1 of Volume 3 of this report gives the staff's regulatory analysis of those issues identified for the passive plant designs. These issues have been addressed in Commission papers SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolution and Advanced Light-Water Reactor (ALWR) Design," and SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety System in Passive Plant Designs," dated April 2, 1993 and March 28, 1994, respectively.

In SRMs dated July 27, 1993, and June 30, 1994, the Commission provided its decisions on SECY-93-087 and SECY-94-084 as they apply to passive plant designs. These issues are considered fundamental to agency decisions on the acceptability of the ALWR designs. The staff will ensure satisfactory implementation of Commission guidance regarding these matters during its review of individual applications for final design approval and design certification.

There are no open issues pertaining to the Requirements Document for passive plant designs.

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PREFACE

This safety evaluation report (SER) (Volume 3) documents the review by the U.S. Nuclear Regulatory Commission (NRC) staff of the 13 chapters of Volume III of the Electric Power Research Institute's (EPRI's) Advanced Light Water Reactor (ALWR) Utility Requirements Document (hereafter referred to as the "passive Requirements Document"). Volume 1, which contains the program summary of the NRC review of Volumes I, II, and III of the ALWR Utility Requirements Document, also contains the references cited and the abbreviations used in this SER.

Each chapter of the passive Requirements Document defines the ALWR Utility Steering Committee's requirements for the design of evolutionary plants. These requirements apply to boiling-water reactors (BWRs) and pressurized-water reactors (PWRs), which will be rated at approximately 600 megawatts-electric.

The design criteria specified by EPRI are intended to ensure the EPRI's policy statements discussed in Volume 1 of the ALWR Utility Requirements Document are met. These policy statements are discussed in Section 1.3 of Volume 1 of this report. They include consideration of simplification, design margin, human factors, safety, regulatory stabilization, standardization, use of proven technology, maintainability, constructability, quality assurance, economics, protection against sabotage, and environmental effects.

The format of each chapter of this SER follows that of the corresponding chapter of the passive Requirements Document as closely as possible. Unless otherwise noted, references to sections of the passive Requirements Document pertain to that chapter.

Outstanding Issues

In the August 1993 draft safety evaluation report documenting its review of the passive Requirements Document, the staff identified two types of issues for which additional information was required before the staff could reach a final conclusion. The staff considered these issues to be outstanding. These issues fell into one of two categories: (1) open issues that had to be resolved before the staff could complete its review of the passive Requirements Document or (2) confirmatory issues for which the staff would ensure that EPRI met its commitments to revise the passive Requirements Document.

There are no open issues remaining on the Requirements Document for passive plant designs. To provide continuity of the review, the open and confirmatory items identified in the DSER are listed in Section 1.4 of each chapter.

During its review of the passive Requirements Document, the staff identified items that were inadequately addressed by EPRI or were issued that could not be addressed generically. These items will have to be resolved during the staff's review of a vendor- or utility-specific application (i.e., an application for final design approval and design certification (FDA/DC) or a combined construction permit and operating license (combined license). They are listed in Section 1.5 of each chapter.

As discussed in Section 1.2 of Volume 1 of this report, the Requirements Document has no legal or regulatory status and is not intended to demonstrate complete compliance with the Commission's regulations, regulatory guidance, or policies. It is not intended to be used as a basis for supporting FDA/DC for a specific design, not is it to be used to substitute for any portion of the staff's review of future applications for FDA/DC. Specifically, satisfactory resolution of the items identified in 1.5 of each chapter for a vendor- or utility-specific application will not, by itself, support a finding that the application complies with the Commission's regulatory requirements. The staff will perform a complete licensing review of these applications using NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," and other appropriate Commission guidance. Satisfactory resolution of the vendor- or utility-specific items constitutes only one portion of the staff's review.

Availability

Copies of this report are available for inspection at the NRC Public Document Room, 2120 L Street, N.W., Washington, DC 20555.

The NRC project manager for the staff's review of EPRI's ALWR Utility Requirements Document is J. H. Wilson. He may be contacted by calling (301) 504-1118 or by writing to: Associate Directorate for Advanced Reactors and License Renewal, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

CHAPTER 2, "POWER GENERATION SYSTEMS"

1 INTRODUCTION

This chapter of the SER documents the NRC staff's review of Chapter 2, "Power Generation Systems," of the passive Requirements Document through Revision 5. Revision 5 of the passive Requirements Document was submitted to the NRC by letter dated October 1, 1993. Chapter 2 was prepared, under the project direction of EPRI and the ALWR Utility Steering Committee, by Bechtel Power Corporation; Duke Power Company; General Electric Company; MPR Associates, Inc.; S. Levy, Incorporated; Science Applications International Corporation; Westinghouse Electric Corporation; and EPRI.

1.1 Review Criteria

Section 1 of Volume 1 of this report describes the review criteria used by the staff during its review of Chapter 2 of the passive Requirements Document.

1.2 Scope and Structure of Chapter 2

Chapter 2 of the passive Requirements Document defines the ALWR Utility Steering Committee's requirements for the design of the systems used to generate power at ALWRs in which passive safety systems will be used.

The key topics addressed in the Chapter 2 review include main/extraction steam system, feedwater and condensate system, chemical addition system, condensate makeup purification system, auxiliary steam system, and PWR backup feedwater system.

1.3 Policy Issues

During its review of Chapter 2 of the passive Requirements Document, the staff did not identify issues that involve policy questions for the technical areas discussed in this chapter, other than those already identified in the Commission papers listed in Appendix B to Chapter 1 of this report.

1.4 Outstanding Issues

The DSER for Chapter 2 of the passive Requirements Document contained the following open issue:

Open Issue

P.2.0-1 safety valve design (3.4)

Confirmatory Issues

None

The final disposition of the open issue is discussed in detail in the appropriate section of this chapter, as indicated by the parenthetical notation following the issue. The open issue identified in the DSER for Chapter 2 has been resolved.

1.5 Vendor- or Utility-Specific Items

The DSER for Chapter 2 listed the following vendor- or utility-specific items. The final disposition of each item is discussed in the appropriate section of this chapter, as indicated by the parenthetical notation following each item.

P.2.V-1 turbine bypass system flow capacity (3.2)
P.2.V-2 attachment loads for PWR safety and relief valves (3.4)
P.2.V-3 design adequacy of side stream polisher (4.3)

The following vendor- or utility-specific items remain and will be addressed during the staff's review of an individual application for FDA/DC. The staff's position on each item is provided in the appropriate section of this chapter, as indicated by the parenthetical notation following each item.

P.2.V-1 compliance with SRP Section 3.9.3 (3.4)
P.2.V-2 design adequacy of side stream polisher (4.3)

2 COMMON REQUIREMENTS

2.1 General Requirements

EPRI states that the design of systems covered by Chapter 2 of the passive Requirements Document will meet the overall requirements of Chapter 1. These systems include the main/extraction steam, feedwater and condensate, chemical addition, condensate makeup purification, auxiliary steam, and PWR backup feedwater systems.

2.2 Specific Requirements

2.2.1 Valves

Section 2.2.1 of Chapter 2 of the passive Requirements Document states that Section 12.2 of Chapter 1 specifies valve and valve actuator requirements applicable to the entire passive ALWR. These requirements provide substantial guidance to plant designers aimed at minimizing and simplifying the valving throughout the plant, including power generation systems, and include

- minimizing the number of valves and valve actuators
- minimizing the number of types, sizes, and models of valves and valve actuators
- use of valves, and actuators of proven design, with emphasis on maintainability, durability, and simplicity
- designing valve packing to reduce packing leakage

The following requirements in Section 2.2.1 of Chapter 2 are specifically applicable to valves in the power generation systems:

- Valves exposed to vacuum should be specified to include two or more of the following features:
  - packless (bellow) seals
  - water seals over the packing
  - chevron-type packing with resilient material
- All valves discharging into the condenser or points of considerably lower pressure must be mounted as close as is practical to the point of discharge, in order to minimize erosion due to flashing flow.

2.2.2 Materials

Section 5 of Chapter 1 specifies material requirements for the ALWR. In particular, Section 5.3.2 contains requirements specifically for the feedwater, steam, and condensate systems, including requirements to minimize use of copper alloys and to use corrosion/erosion-resistant materials (not carbon steel) for piping and components exposed to wet steam or flashing liquid flow. The staff concludes that these requirements for materials in the steam,

condensate, and feedwater systems ensure that system components will be compatible with their intended service conditions and are, therefore, acceptable.

2.2.3 Instrumentation and Control

Section 2.2.3 of Chapter 2 states that the instrumentation and control (I&C) equipment for the power generation systems will meet the requirements of Chapter 10 of the passive Requirements Document. Although Chapter 2 defines functional requirements that will affect the type, range, and location, the actual design requirements are identified in Chapter 10. Controls and displays and their location will also be established by the analyses of functions also required by Chapter 10. The staff has, therefore, documented its review of the I&C requirements in Chapter 2 of the passive Requirements Document in Chapter 10 of this report.

3 MAIN/EXTRACTION STEAM SYSTEM

3.1 System Definition

Section 3.1.1 of Chapter 2 of the passive Requirements Document states that the main/extraction steam system will be designed to

- transport main steam from the steam generator (for PWR) or main steam isolation valve (for BWR) to the high-pressure turbine and to the moisture separator reheater
- transport extraction steam from the high-pressure and low-pressure turbines to the feedwater heaters
- provide steam to the auxiliary steam system (for PWR) and provide steam bypass capability via the turbine bypass system for startup, shutdown, and step-load reduction transients (for BWR)
- provide isolation of the main steamlines in case of a main steamline break (for PWR)
- provide steam to steam jet air ejectors and to gland seals

PWR System Boundaries

Section 3.1.2 of Chapter 2 defines the main/extraction steam system boundaries as consisting of the following for a PWR:

- main steam piping up to, but not including, the turbine stop valves
- hot reheat piping up to, but not including, the reheat stop valves
- extraction steam and cold reheat piping
- turbine bypass system
- moisture separator/reheater
- main steam isolation valves (MSIVs)
- safety valves
- power-operated relief valves (PORVs)

BWR System Boundaries

Section 3.1.2 of Chapter 2 defines the main/extraction steam system boundaries as consisting of the following for a BWR:

- main steam piping downstream of the second MSIV up to, but not including, the turbine stop valves
- hot reheat piping up to, but not including, the reheat stop valves
- extraction steam and cold reheat piping
- turbine bypass system
- moisture separator/heater

The BWR MSIVs are not included because they are addressed in Chapter 3 of the passive Requirements Document. EPRI's requirements for the turbine-generator are given in Chapter 13.

Interfaces

In the DSER for Chapter 2, the staff recommended that the interface with the emergency feedwater system of Chapter 5 of the passive Requirements Document be deleted, as that system will not exist in the passive plant. In Revision 4, EPRI revised the fifth bullet of Section 3.1.3 of Chapter 2 to denote the interface with the backup feedwater system of Section 8 of this chapter.

Section 3.1.3 of Chapter 2 now lists the following systems with which the main/extraction steam system will interface: the BWR reactor coolant system (RCS) and PWR steam generator system (Chapter 3), the turbine-generator system (Chapter 13), the extraction steam system (Section 4), the auxiliary steam system (Section 7), the backup feedwater system turbine-driven pumps (Section 8), the main condenser (Section 4), the process sampling system (Chapter 3), and the BWR radwaste drain system (Chapter 12).

This revision corrects the earlier error and is acceptable.

3.2 Performance Requirements

Section 3.2 of Chapter 2 of the passive Requirements Document defines the performance requirements for steam bypass and relief capability, MSIVs, and MSIV bypass lines.

Section 3.2.1.1.3 of Chapter 2 states that all of the turbine bypass system flow will be directed to the condenser in order to conserve secondary water inventory. In the DSER for Chapter 2, the staff noted its concern about those times when the main condenser will not be available. The staff stated that applicants referencing the passive Requirements Document should consider allowing the release of uncontaminated steam through the steam bypass and relief system at settings below which the safety/relief valves operate. This procedure would be consistent with the rationale to minimize safety valve actuations and to conserve secondary water inventory.

In its letter dated September 29, 1992, EPRI responded that such an alternate pathway was unnecessary for the BWR for the following reasons. If the main condenser is not available during power operations, the reactor will be scrammed and residual decay heat will be absorbed by the isolation condenser. Another alternate steam release pathway was via the reactor steam relief valve to the suppression pool. For the PWR, if the main condenser is not available, an alternate steam discharge pathway will be via the steam generator PORVs, which will discharge directly to the atmosphere (see Section 3.4.3.3 of Chapter 2 of the passive Requirements Document). These are in addition to the safety valves described in Section 3.4.3.2 of Chapter 2.

This addresses the staff's concern that the PWR turbine bypass steam flow have an alternate path in addition to the condenser, available for discharge.

Section 3.2.1.3.1 of Chapter 2 states that the turbine bypass system total flow capacity will be 33 percent of the full turbine steam flow at full-load

steam pressure. The staff noted in the DSER that this requirement might conflict with Section 3.2.1.2.1 of Chapter 2, which requires that the turbine bypass system total flow capacity be sufficient to eliminate challenges to the steam generator PORVs during reactor trip from a full-power transient or turbine trips without reactor trip from 100-percent power. It was not clear if bypassing 33 percent of the full turbine steam flow could eliminate PORV challenges for these trips. Therefore, the staff stated that applicants referencing the passive Requirements Document would need to clarify this matter.

In a letter dated September 29, 1992, EPRI stated that the requirement of 33-percent bypass flow in Section 3.2.1.3.1 applies to the BWR plant only. There is no conflict with Section 3.2.1.2.1, since this section applies to the PWR plant only. The staff agrees with EPRI's response. Therefore, this is no longer considered to be a vendor- or utility-specific item.

In the DSER, the staff noted that Table T.3-1 referenced in Section 3.2.2 of Chapter 2, which provides the MSIV design requirements, should be cited as T.2.3-1 and stated EPRI should correct this error. In Revision 4, EPRI revised Section 3.2.2 to correctly reference Table 2.3-1.

3.3 System Features

Section 3.3 of Chapter 2 of the passive Requirements Document defines requirements for system arrangement, system pressure drops and volumes, steam piping drains, and chemistry sampling connections.

3.4 Component Features

Section 3.4 of Chapter 2 of the passive Requirements Document gives the requirements for main/extraction steam system components.

Main Steamline Classification

In the DSER, the staff reported that Section 3.4.1.5 of Chapter 2 required that the main steamline from the seismic restraint on the outboard side of the outmost MSIVs to and including the turbine main steam stop valves meet seismic Category II requirements. The seismic Category II requirements were not completely acceptable to the staff for these components because, in part, they allow the use of the seismic experience database, which has not been accepted by the staff for the design of piping. In response to the DSER, in Revision 3 of the passive Requirements Document, EPRI revised Section 3.4.1.5 to agree with staff positions on these components that resulted from the elimination of the BWR main steam isolation valve leakage control system. This issue is discussed in Section 2.3.1 of Appendix B to Chapter 1 of this report.

Extraction Steam and Cold Reheat Piping

Section 3.4.2.3 of Chapter 2 specifies that the extraction steam and cold reheat piping material be of corrosion-resistant materials meeting the requirements of Chapter 1, Section 5.3.2. Carbon steel must not be used.

The staff has reviewed the materials requirements for extraction steam and cold reheat piping in Section 5.3.2 of Chapter 1 of the passive Requirements

Document and concludes that they ensure that the piping will be compatible with its expected service conditions and are, therefore, acceptable.

Safety Valves (PWR)

Section 3.4.3.2.2 of Chapter 2 states that the safety valves must be of a design proven to consistently open fully, at a pressure within acceptable limits around the set pressure, during operability tests. In the DSER for Chapter 2, the staff stated that the design requirements and rationale for the safety valves were based on their functioning during operability tests and identified this as an open issue. In Revision 4, EPRI revised Section 3.4.3.2.2 by adding the following statement: "The design of the safety valves shall be based on their being able to function at accident conditions." EPRI also revised the rationale to state: "This requirement is intended to assure that these valves are able to function in harsh environments to mitigate accidents and to minimize the possibility of plant shutdown and/or safety valve maintenance because of a safety valve not opening in the specified pressure range during an operability test." Chapter 1 of the passive Requirements Document contains requirements to design ASME Class 2 and 3 valves in accordance with ASME Code, Section III, using acceptable design-basis accident loading combinations. Therefore, ASME Code, Section III, Subsections NC/ND-7512, "Safety Valve Operating Requirements," will also be implemented. On the basis of the above additions to the requirements of Chapter 2 plus the existing requirements in Chapter 1, the staff concludes that ALWR main steam safety valves in PWRs will be designed to maintain their structural integrity and functionability under accident loading conditions. These changes are acceptable and the open issue is closed.

Safety and Relief Valves (PWR)

By letter dated May 17, 1991, the staff requested that Sections 3.4.3.2.4 and 3.4.3.3.3 of Chapter 2, which discuss attachment loads for PWR safety and relief valves, respectively, be revised to delete a reference to American National Standards Institute/American Society of Mechanical Engineers (ANSI/ASME) B.31.1, Appendix 2, "Non-Mandatory Rules for the Design of Safety Valve Installations," and that the reference be replaced with a requirement to design such installations in accordance with the acceptance criteria in Item II.2 of Standard Review Plan (SRP) Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures" (NUREG-0800). In Revision 2 of the passive Requirements Document, the reference to B.31.1 was deleted from both of the above sections in Chapter 2, but no reference was made to SRP Section 3.9.3. However, in Table B.1-2 in Appendix B to Chapter 1 of the passive Requirements Document, EPRI makes the commitment that the ALWR will comply with SRP Section 3.9.3 unconditionally. The staff concludes that this is an acceptable commitment and will review individual applications for both BWR and PWR FDA/DC by implementing the guidance in Item II.2 of SRP Section 3.9.3.

Moisture Separator/Reheaters

Section 3.4.4.4 of Chapter 2 requires that corrosion/erosion-resistant materials meeting the requirements of Chapter 1, Section 5.3.2, of the passive Requirements Document, be used for all components exposed to wet steam or

flashing liquid flow where operating experience has shown that significant erosion could occur. In addition, Section 3.4.4.6 of Chapter 2 requires that reheater tubes be made of ferritic stainless steel.

The staff concludes that these requirements are acceptable because flashing liquid flow and wet steam service have been shown to be an erosion problem. Ferritic stainless steel tube material is corrosion/erosion resistant and is compatible with service in both BWRs and PWRs.

3.5 Instrumentation and Control

Section 3.5 of Chapter 2 of the passive Requirements Document requires instrumentation and control (I&C) equipment for the main/extraction steam system, including

- heat balance instrumentation
- turbine bypass system valve interlocks and trips
- steamline break detection (PWR)
- power-operated relief valves (PWR)
- main steam isolation valves (PWR)
- main steam isolation valve bypass valves (PWR)
- turbine main steam bypass valves
- main steam drain pot valves
- heater extraction steamline isolation and power-assisted non-return valves
- moisture separator reheating steam shutoff, control, and purge valves

The staff's evaluation of the design requirements for this I&C equipment is given in Chapter 10 of this report.

3.6 Maintenance

Section 3.6 of Chapter 2 refers to the general maintenance requirements in Chapter 1 of the passive Requirements Document. EPRI states that the requirements in this section of Chapter 2 are intended to ensure that maintenance of all main steam system components can be accomplished quickly and safely. Section 3.6 requires that the following specific provisions be addressed during the design phase:

- adequate work space and ease of access to equipment and components
- adequate space and lifting provisions for removal of valve components
- adequate space for laydown of equipment

- moisture separator reheater pull fixtures specifically designed for each location

- physical layout to ensure safety of personnel during maintenance

The staff concludes that these requirements are acceptable because they represent reasonable provisions to address conditions that have hindered maintenance at operating plants.

3.7 Conclusion

The staff concludes that, with the exception of the issues to be addressed by applicants referencing the passive Requirements Document, as noted above, the design requirements in Chapter 2 of the passive Requirements Document for the main/extraction steam system are in general agreement with regulatory requirements or staff guidelines and are, therefore, acceptable.

4 FEEDWATER AND CONDENSATE SYSTEM

4.1 System Definition

The feedwater and condensate system will be designed to return high-quality feedwater from the condenser hotwell to the steam generators (for PWR) or reactor vessel (for BWR). Section 4.1.3 of Chapter 2 of the passive Requirements Document listed the systems with which the feedwater and condensate system will interface. One of the paragraphs in this section stated that the reactor coolant system will receive feedwater from the feedwater and condensate system. In the DSER, the staff noted that this paragraph applied to BWRs only and recommended that it be so specified. Also, the auxiliary steam system, described in Section 7 of Chapter 2, is another interface with the feedwater and condensate system. In Revision 4, EPRI revised Section 4.1.3 as follows:

(1) Changed the fifth bullet to read: "The reactor coolant system (BWR) and steam generator system (PWR) (Chapter 3) receive feedwater from the feedwater and condensate systems."

(2) Added a tenth bullet that reads: "The auxiliary steam system (PWR), (Section 7) may receive condensate makeup from the feedwater and condensate system."

The staff concludes that the changes are acceptable.

4.2 Performance Requirements

Section 4.2 of Chapter 2 of the passive Requirements Document specifies that this system will be designed with the capability of automatically providing the required flow to the steam generators (PWR) or reactor vessel (BWR) during startup, shutdown at power levels up to, and including, rated load and during the plant design transients without interruption of operation or damage to equipment. Section 4.2.1.2 states that the capability to operate the plant at full power will be maintained if one condensate pump should fail. Section 4.2.1.9 states that loss of a single feedwater pump (BWR), booster/main feedwater pump assembly (PWR), heater drain pump (BWR), or condensate pump in a multiple pump system should not result in trip of the turbine-generator or a reactor trip.

4.3 System Features

Demineralizer/Condensate Polisher

Section 4.3.9.1.1 of Chapter 2 of the passive Requirements Document requires that the demineralizer/condensate polisher maintain water quality suitable for long-term power operation, startup, shutdown, and extended outages. In a passive ALWR plant, properly designed condensate polishers will maintain water chemistry within specified limits, assuming a condenser tube leak of 0.001 gpm during continuous operation and 0.1 gpm during an orderly unit shutdown not longer than 8 hours. In addition, the polisher system will provide adequate cleanup function during plant heatup and low-power operation. No ion exchange resin regeneration system will be provided.

In a PWR, a side stream condensate polisher with deep-bed, mixed-resin ion exchangers will be used to maintain feedwater chemistry within specified limits. Section 4.3.9.2.2 states that, in part, if the system is sized for less than full condensate flow, it should be capable of handling at least one-third of rated condensate flow. At a site using seawater cooling, a full condensate flow rate system may be required. The sizing of the polisher is intended to protect the steam generators and other secondary-side components from corrosion resulting from poor-quality makeup water. If a full-flow system will not be provided, the design and arrangement should include provisions for the possible future installation of full-flow capability. However, the staff is concerned that a side stream condensate polisher may not be adequate for flow control if a full condensate flow rate system is needed. The staff will review this matter on a plant-specific basis.

In a BWR, a deep-bed, mixed-resin, full-flow ion exchange demineralizer will be used. The total condensate flow will be processed through a full-flow filter to remove particulates from the condensate stream before it enters the deep-bed condensate demineralizer. The system will maintain feedwater chemistry within specified limits. The water chemistry in the system will be further controlled by deaerating the condensate during startup and during normal plant operation. In addition, there will be a provision for injecting chemicals into the condenser for biofouling control. EPRI states that the condensate polisher, ion exchange demineralizer, and filter system will be designed to comply with the general requirements in Chapter 1 of the passive Requirements Document.

To reduce the potential for resin fines being transported into the steam generators (PWR) or reactor pressure vessel (BWR), Section 4.3.9.1.3 requires that resin traps be installed downstream of each ion exchanger to remove resin fines (in the event of a failure of an internal screen in the ion exchanger). In Section 12.9.3.14 of Chapter 1, EPRI states that personnel exposure will be minimized during resin trap cleaning by locating resin traps outside the ion exchanger enclosure and by providing remote backwash capability for resin traps. The staff concludes that this design provision is acceptable.

Capability for Handling Radioactivity

Section 4.3.12 of Chapter 2 states that to reduce the amount of liquid radioactive water that must be processed, BWR condensate, feedwater, and heater drain pump seal leakage will be drained to the condenser hotwell. Any primary-to-secondary PWR leakage will be removed via the steam generator blowdown demineralizers. These demineralizers, as well as the condensate polishers, will be located in areas in which temporary shielding can be installed if necessary.

4.4 Component Features

Section 4.4.1 of Chapter 2 of the passive Requirements Document states that the provisions of Chapter 1, requiring that only components that have been proven in comparable service be used, must be followed. Also, to eliminate overspeed trips on the condensate pumps and the main feed pumps, the design pressure for all components downstream of the pumps will be equal to or higher than the discharge pressure of the respective pumps, assuming no flow (shutoff

head developed across pumps). Finally, portions of the feedwater and condensate system that will be under vacuum conditions during low power and startup must be designed to prevent air inleakage and to maintain acceptable water chemistry.

Condenser

Section 4.4.3 of Chapter 2 requires that the condenser be designed in accordance with Heat Exchanger Institute standards. The condenser will have two or more parallel circulating water flow paths. Tubing must be of commercially available lengths. The design must not preclude shop prefabrication.

In addition, the following requirements must be met:

- The condenser tube will be made of Type 304L stainless steel for fresh water with chloride levels below 200 parts per million (ppm). For higher chloride levels of up to 500 ppm, Type 316L stainless steel tubing will be used. A higher grade of stainless steel (such as 904L or AL-6X) must be used if chloride levels are between 500 and 800 ppm. Titanium tubing will be used for brackish or salt water containing high concentrations of dissolved solids (1000 ppm) or chlorides (more than 800 ppm) or water contaminated by sewage discharges. Titanium tubing may be used for any water condition if the plan designee's studies show it is optimum as a standard design.

- Stainless steel tubing material must not be thinner than 22 British wire gauge (BWG). Titanium tubing material must not be thinner than 23 BWG. Protection against impingement will be provided. Support plates will be designed to minimize tube vibrations.

- Provisions for chemical injection into the condensate for biofouling control must be included in accordance with site-specific requirements and applicable regulations.

- Tubes must be protected from pitting when the condenser is shut down.

- Tube sheets will be specified as follows:

  - For Type 304L stainless steel tubes, Type 304L stainless-clad carbon steel tube sheets will be used.

  - For Type 316L stainless steel tubes, Type 316L stainless-clad carbon steel tube sheets will be used.

  - For higher grade stainless steel tubes, stainless-clad carbon steel tube sheets will be used.

  - For titanium tubing, titanium-clad carbon steel tube sheets will be used.

- Double tube sheets or welded tube-to-tube-sheet joints will be provided.

- The formation of corrosion products and the loss of condenser materials will be minimized by eliminating erodible steel surfaces and/or by using materials other than carbon steel.

- Leak detection trays will be included at all tube-to-tube-sheet interfaces. Early leak detection will be provided at tube sheet trays and in each hotwell section. The hotwell will be divided into sections so that leaks can be detected and located.

The staff concludes that these requirements are acceptable because corrosion-resistant materials are to be used in the construction of the condenser. Also, leak detection trays are required for early leak detection.

Feedwater Heater and Deaerator

Section 4.4.4.1 of Chapter 2 specifies that feedwater heater tubes be made of Type 304L stainless steel with carbon steel tube sheets. Tube-to-tube sheet joints must be welded.

The staff concludes that these requirements are acceptable because Type 304L stainless steel tubes have shown good resistance to the type of physical and chemical attack common in the power plant feedwater system environment.

Regulating Valves

Section 4.4.8.1 of Chapter 2 requires that appropriate stainless steel materials be specified for all valve bodies and internal components for regulating applications in the feedwater and condensate system, including drain valves on heaters. Design features will facilitate inspection, maintenance, and replacement, as required, of regulating valve internal components or valve seats.

The staff concludes that these requirements are acceptable because stainless steel will provide increased resistance to cavitation and erosion damage.

4.5 Instrumentation and Control

Section 4.5 of Chapter 2 of the passive Requirement Document requires instrumentation and control provisions for the feedwater and condensate system, including

- heat balance instrumentation
- reactor and turbine trips
- level controls
- turbine water induction prevention controls
- condenser hotwell level control
- feedwater string isolation valves
- deaerator storage tank level control (PWR)
- feedwater heater drain controls
- pump trips

4.6 Maintenance

Section 4.6 of Chapter 2 refers to the general maintenance requirements in Chapter 1 of the passive Requirements Document. EPRI states that the requirements in Chapter 2 are intended to ensure that maintenance can be performed on all components of the feedwater and condensate system quickly and safely. Section 4.6 of Chapter 2 requires that the specific provisions for ease of access, adequacy of work space, and laydown areas be addressed during the design phase.

4.7 Conclusion

The staff concludes that the EPRI requirements for the design of the feedwater and condensate system are in general agreement with SRP Section 10.4.6, "Condensate Cleanup System," and SRP Section 10.4.7, "Condensate and Feedwater System," and are acceptable.

5 CHEMICAL ADDITION SYSTEM

5.1 System Definition

The chemical addition system will be designed to add liquid (for PWR) or gaseous (for BWR) chemicals as necessary to maintain condensate, feedwater, and the off-gas (for BWR) system chemistry within the required limits.

5.2 Performance Requirements

Section 5.2.1 of Chapter 2 of the passive Requirements Document requires that the chemical addition systems for PWRs maintain water quality for long-term operation during all plant conditions. The system will have sufficient capacity to continuously inject chemicals for 24 hours to limit the need to replenish chemicals. Specific feedwater chemistry requirements are given in Chapter 3 of the passive Requirements Document.

Section 5.2.2 of Chapter 2 requires that the gas addition system for BWRs be capable of maintaining the required gas concentrations at all power levels above 30 percent of full power. Hydrogen addition is part of the industry's BWR Hydrogen Water Chemistry (HWC) Program (in Chapter 1 of the passive Requirements Document). The system will be capable of providing the prescribed gas concentrations 90 percent of operating time.

5.3 System Features

PWR

Section 5.3.1 of Chapter 2 specifies, for PWRs, that separate and identical equipment and injection tubing will be provided in the chemical addition system for adding hydrazine and ammonia or morpholine. The system will consist of the chemical addition tanks, pumps, piping, instrumentation, and addition points on the feedwater and condensate system. EPRI requires that the system be designed to inject suitable amounts of hydrazine and ammonia or morpholine during plant operation and during plant layups.

BWR

Section 5.1.2 of Chapter 2 specifies, for BWRs, that the chemical addition system will consist of gas generation and/or storage facilities, piping, flow metering, instrumentation, and addition points on the feedwater and off-gas systems. Section 5.3.2 of Chapter 2 specifies that separate gas injection tubing will be provided for (1) injecting oxygen into the condensate, (2) injecting hydrogen into the feedwater, and (3) adding oxygen to the off-gas system.

5.4 Component Features

Sections 5.4.1 and 5.4.2 of Chapter 2 require that the chemical addition pumps in PWRs be fabricated from Type 316 stainless steel and the chemical addition tanks be fabricated from Type 304 stainless steel. Section 5.4.3 requires that the piping and tubing for the gas addition systems in BWRs be made of Type 316 stainless steel.

The staff concludes that these requirements are acceptable because the specified materials are compatible with the intended service.

5.5 Instrumentation and Control

For PWRs, Section 5.5 of Chapter 2 requires automatic control of hydrazine and ammonia addition pumps and level switches with low-level alarms and pump trip controls on chemical addition tanks.

For BWRs, Section 5.5 of Chapter 2 requires automatic control of condensate oxygen addition flow, feedwater hydrogen injection flow, and off-gas oxygen addition flow.

5.6 Maintenance

Section 5.6 of Chapter 2 refers to the general requirements in Section 8 of Chapter 1 of the passive Requirements Document. Section 5.6 of Chapter 2 requires that chemical addition skids be designed for ease of maintenance and quick replacement of individual components.

5.7 Conclusion

The staff concludes that the EPRI requirements for the design of the chemical addition system for the PWR are in general agreement with SRP Section 5.4.2.1, Branch Technical Position MTEB 5-3, "Monitoring of Secondary Side Water Chemistry in PWR Steam Generators." For the BWR, the design of an onsite hydrogen generator facility, if provided, will follow the guidance in EPRI's report NP-5283-SR-A(1987), "Guidelines for Permanent BWR HWC Installations." These requirements do not conflict with other regulatory requirements and are, therefore, acceptable.

6 CONDENSATE MAKEUP PURIFICATION SYSTEM

6.1 System Definition

The condensate makeup purification system will be designed to treat the raw makeup water and store the treated water for filling, flushing, and providing makeup water for the feedwater and condensate system. In addition, the system will provide purified water to the primary water storage system for the PWR plant. The complete system will consist of a demineralizer for removing ionic impurities, a vacuum degasifer for removing dissolved oxygen, a demineralizer water storage tank for sampling, and a condensate storage tank. In some cases, additional equipment may be needed to filter, clarify, and soften the water depending on the quality of the makeup water.

Interfaces

Section 6.1.3 of Chapter 2 of the passive Requirements Document describes the system interfaces with the raw water makeup system (Chapter 9), the feedwater and condensate system (Section 4 of this chapter), the PWR chemical and volume control system (CVCS) (Chapter 3), the process sampling system (Chapter 3), and the fuel pool (Chapter 7). The staff finds that the cited chapters address the system interfaces.

6.2 Performance Requirements

Section 6.2 of Chapter 2 requires that the condensate makeup purification system provide condensate makeup water of a quality and quantity suitable for long-term plant operation, including power operation, startup, shutdown, extended outage, and off-chemistry conditions. PWR system capacity will be based on the maximum steam generator blowdown rate and miscellaneous condensate requirements. BWR system capacity will be based on auxiliary steam makeup requirements during plant startup and miscellaneous condensate requirements.

6.3 System Features

Section 6.3 of Chapter 2 specifies features of the condensate makeup purification system that are required for a typical site with good water quality. Plant designers are required to review the quality of raw water for each specific site and provide adequate equipment, as required, for pretreatment. Condensate storage and demineralized water storage tanks will be designed to maintain water purity and exclude oxygen. Section 6.3.2.2 of Chapter 2 requires that the demineralizer system be designed with two 100-percent-capacity trains and include such features as

- strainers in waste lines to eliminate resin carryover during backwash
- use of inert resin in mixed-bed vessels
- full-flow recirculation
- resin regeneration
- sight glasses for viewing resin levels in mixed-bed vessels
- resin traps downstream of each demineralizer vessel

6.4 Component Features

Section 6.4 of Chapter 2 gives specific requirements for components of the condensate makeup purification system, including the demineralizer, vacuum degasifier, demineralized water storage tank, and condensate storage tanks.

Demineralizer

The demineralizer will include cation, anion, and mixed-bed units. A decarbonator may also be provided depending on the alkalinity of the makeup water. Depending on site-specific raw water quality, the designer may specify a different demineralizer arrangement based on an evaluation of site-specific conditions.

Section 6.4.1.4 of Chapter 2 requires the following:

- Demineralizer vessels must be constructed of lined carbon steel.
- Demineralizer skid piping must be constructed of polypropylene-lined carbon steel.
- Dilute acid piping material must be Alloy 20.
- The demineralizer waste tank must include a liner that can withstand the corrosive effects of the regenerated waste over the complete range of expected pH values and chemical concentrations. This tank will include provisions for chemical neutralization.
- Demineralizer waste piping may be constructed of Alloy 20 or other corrosion-resistant material such as polyethylene or polypropylene-lined steel. This piping will be routed above grade so that piping leaks can be detected.

The staff concludes that these requirements are acceptable because the materials specified are standard for demineralizer systems.

Vacuum Degasifier

The vacuum degasifier will be of the packed spray tower type design with makeup water injected at the top of the bed through a distribution system. Two vacuum pumps will be provided to maintain system vacuum.

Section 6.4.2.2 of Chapter 2 requires that the degasifier vessel be constructed of rubber-lined carbon steel. All piping valves and fittings should be made of Type 304 stainless steel.

The staff concludes that these requirements are acceptable because the materials specified are standard materials of construction for the vacuum degasifier.

Demineralized Waste Storage Tank

A demineralized waste storage tank (DWST) will be provided to sample the quality of water before it is released to the plant. The capacity of this tank will be based on the design flow rate of the demineralizer and the makeup requirements of the condensate storage tank.

Section 6.4.3.2 of Chapter 2 requires that the DWST be constructed of stainless steel. A stainless steel floating cover on the DWST is recommended to minimize air ingress.

The staff concludes that these requirements are acceptable because the materials specified are standard materials of construction for the DWST.

Condensate Storage Tanks

The minimum capacity of the condensate storage tanks will be based on the maximum condensate usage during startup (e.g., for the PWR, maximum steam generator blowdown level x startup duration) plus a 100-percent margin. A minimum of two pumps will be provided for recycling condensate back to the degasifier or forwarding it to the locations where it will be used in the plant.

Section 6.4.4.2 of Chapter 2 requires that the condensate storage tanks be constructed of stainless steel. Stainless steel floating covers on the tanks are recommended to minimize air ingress.

The staff concludes that these requirements are acceptable because the materials specified are the standard materials of construction for the condensate storage tank.

6.5 Instrumentation and Control

Section 6.5 of Chapter 2 requires that the condensate makeup purification system be controlled and monitored from local control panels. Provisions will be made for manual control as backup to the automatic or semiautomatic control. Table 2.6-1 of Chapter 2 gives the parameters that should be maintained throughout the demineralizer system. Section 6.5.6 of Chapter 2 requires that provisions be made for appropriate automatic trips and/or isolations to protect against chemical intrusion (i.e., when limits of silica, sodium, or conductivity in demineralizer effluent are exceeded).

6.6 Maintenance

Section 6.6 of Chapter 2 requires that the condensate makeup sampling system be located near the processing equipment within an environmental enclosure to protect operators and equipment from the adverse effects of temperature, humidity, chemical or steam leaks, and local noise.

6.7 Conclusion

The staff concludes that the design requirements for the condensate makeup purification system do not conflict with the regulatory requirements in SRP Section 9.2.3, "Demineralized Water Makeup System," and are acceptable.

7 AUXILIARY STEAM SYSTEM

7.1 System Definition

The auxiliary steam system will be designed to supply low-pressure nonradioactive steam to various plant components. For PWRs, the system will be supplied by nonradioactive service boiler(s) or reduced-pressure, potentially radioactive main steam. Strictly nonradioactive services (e.g., station heating steam) must always be supplied with nonradioactive steam. For BWRs, the system will be supplied with nonradioactive steam from service boiler(s).

7.2 System Interfaces

The auxiliary steam system will interface with the following systems:

- main steam system (PWR) (Section 3 of this chapter)
- deaerator (PWR) in the feedwater and condensate system (Section 4 of this chapter)
- turbine gland sealing system (Chapter 13)
- PWR chemical and volume control system (Chapter 3)
- liquid radwaste system evaporator, if used (Chapter 12)
- space and hot water heating system (Chapter 9)
- BWR combustible gas control (inerting) system (Chapter 5)
- process sampling system (Chapter 3)

7.3 Performance Requirements

Section 7.3.1 of Chapter 2 of the passive Requirements Document states that the auxiliary steam system will have no safety-related function and will provide steam for the following:

- deaerator pegging (PWR)
- turbine gland sealing for the PWR and as needed for the BWR
- liquid nitrogen evaporation for inerting the containment (BWR)
- radioactive waste evaporator/concentrator and/or evaporator of the PWR borated waste processing subsystem, if included
- building heating when needed
- miscellaneous components

The system will provide the plant with the operational flexibility necessary to supply the required steam loads during all modes of plant operation.

Section 7.3.3 of Chapter 2 states that the system will be designed to maintain steam quality consistent with the requirements of the feedwater and condensate system.

7.4 System Features

Section 7.4 of Chapter 2 requires that, for the PWR, the auxiliary steam system be supplied with steam from a package steam boiler for plant startup and from the steam system for normal operation. For the BWR, the nonradioactive auxiliary steam will be supplied from a service boiler.

At operating plants, poorly designed interfaces between the auxiliary steam system and radioactive components have resulted in the contamination of the auxiliary steam system by radioactive steam or liquid. Section 7.4.3 of Chapter 2 states that the auxiliary steam system will be designed so that radioactive steam or liquid will not cause contamination of the system.

7.5 Component Features

Section 7.5 of Chapter 2 states that only components with proven service will be used in the auxiliary steam system. Section 7.5.5 of Chapter 2 states that the auxiliary steam boiler system will consist of two condensate collecting pumps, two boiler makeup pumps, and two boiler feed pumps. Each pump will be designed for 100-percent flow and will have a constant recirculation line to meet minimum flow requirements.

7.6 Maintenance

Section 7.6 of Chapter 2 refers to the general maintenance requirements in Section 8 of Chapter 1.

7.7 Conclusion

Since the auxiliary steam system will have no safety-related function, the NRC has no regulatory requirements for the system. However, the design requirements for the system generally reflect good engineering practice and are acceptable.

8 PWR BACKUP FEEDWATER SYSTEM

8.1 System Definition

Section 8.1.1 of Chapter 2 of the passive Requirements Document states that the backup feedwater system will be designed to supply feedwater to the steam generators for removing decay heat from the reactor coolant system (RCS) when the main feedwater system is unavailable. The system will not be a safety-grade system and will serve as a first line of defense for such events. The safety-grade passive residual heat removal (RHR) system will perform the safety-related decay heat removal function when no feedwater is available.

The feedwater and condensate system will provide water to the backup feedwater system (through the deaerator storage tank) and will control flow through the startup feedwater valves. If normal feedwater is lost, the backup feedwater system will refill the steam generators and support RCS cooldown by removing heat through the steam generators.

8.2 Performance Requirements

Section 8.2 of Chapter 2 specifies performance requirements for the backup feedwater system to ensure that the requirements in Section 3.4.3.2.1 of Chapter 3 of the passive Requirements Document can be met without actuating the passive RHR system. Also, Section 8.2.2 of Chapter 2 contains requirements to ensure that mass and energy input to the containment resulting from flow of backup feedwater to the affected steam generator following a main steamline break is accounted for in the containment design. EPRI prohibits automatic isolation of backup feedwater.

8.3 System Features

Section 8.3.1.1 of Chapter 2 states that the system will meet the requirements for non-safety-grade auxiliary systems in Chapter 3 of the passive Requirements Document. Requirements for system configuration, pumps, recirculation lines, pump suction lines, pump discharge lines, design pressure, and pump steam binding are given in Section 8.3 of Chapter 2.

8.4 Component Features

Section 8.4 of Chapter 2 specifies requirements for pumps and valves in the backup feedwater system that are based on operating experience and that will facilitate maintenance activities.

8.5 Instrumentation and Controls

Section 8.5.1 of Chapter 2 requires the plant designer to analyze each automatic control loop of the backup feedwater system to demonstrate that stable operation will be achieved over the full range of possible operating conditions. The design requirements of Section 8.5.2 specifically prohibit a pump trip on low suction pressure; therefore, suction pressure monitoring and alarm will be provided as an alternative because such trips have caused loss of all feedwater pumps on low suction pressure during pump startup at operating plants.

8.6 Conclusion

EPRI states that the system should be designed to limit RCS pressure rise and to reduce the frequency of core damage for the passive plant risk analysis. The staff concludes that the amount of credit taken for the backup feedwater system must be commensurate with the classification of the system as non-safety grade.

The backup feedwater system will replace the auxiliary feedwater system, which traditionally has been a safety-grade system. However, for the passive ALWR, the system will not be designed to meet safety-grade criteria, and credit will not be taken for it in the SRP Chapter 15 licensing analysis for the design-basis accident. That analysis relies solely on the passive safety systems to demonstrate compliance with the acceptance criteria for various design-basis transients and accidents.

9 CONCLUSION

The staff concludes that the EPRI requirements established in Chapter 2 of the passive Requirements Document for the design of power generation systems do not conflict with current regulatory guidelines and are acceptable. However, by themselves, they do not provide sufficient information for the staff to determine if the design of these systems will be adequate. Applicants referencing the passive Requirements Document will be required to demonstrate compliance with the additional guidance in the SRP (NUREG-0800), or provide justification for alternative means of implementing the associated regulatory requirements.

Therefore, the staff concludes that Chapter 2 specifies requirements that, subject to the resolution of the identified vendor- and utility-specific items, if properly translated into a design in accordance with the NRC regulations in force at the time the design is submitted, should result in a nuclear power plant whose power generation systems will perform as designed and have all the attributes required by the regulations to ensure that there is no undue risk to the health and safety of the public or to the environment. In addition to complying with existing regulations, such a facility would also be consistent with Commission policies on severe-accident protection and public safety goals.

APPENDIX A
DEFINITIONS AND ACRONYMS

Appendix A of Chapter 2 of the passive Requirements Document contains definitions of acronyms. The staff has provided a consolidated list of acronyms in Volume 1 of this report.

CHAPTER 3, "REACTOR COOLANT SYSTEM AND REACTOR NON-SAFETY AUXILIARY SYSTEMS"

1 INTRODUCTION

This chapter of the SER documents the NRC staff's review of Chapter 3, "Reactor Coolant System and Reactor Non-Safety Auxiliary Systems," of the passive Requirements Document through Revision 5. Revision 5 was submitted to the NRC by letter dated October 1, 1993. Chapter 3 was prepared, under the project direction of EPRI and the ALWR Utility Steering Committee, by Bechtel Power Corporation; Duke Power Company; General Electric Company; MPR Associates, Inc.; S. Levy Incorporated; Sargent and Lundy Engineers; Westinghouse Electric Corporation; and EPRI.

1.1 Review Criteria

Section 1 of Volume 1 of this report describes the review criteria used by the staff during its review of Chapter 3 of the passive Requirements Document.

1.2 Scope and Structure of Chapter 3

Chapter 3 of the passive Requirements Document defines the ALWR Utility Steering Committee's requirements for the design of the reactor coolant system and the reactor non-safety-related auxiliary systems for passive ALWR plants. The non-safety-related auxiliary systems include systems that are required for normal operation but are not required to operate for mitigating accidents or for bringing the plant to a safe-shutdown condition. For PWRs, the reactor coolant system and the boron recycle system are included. For BWRs, the reactor coolant system and the reactor water cleanup system are included. Chapter 3 also includes requirements for process sampling.

1.3 Policy Issues

During its review of Chapter 3 of the passive Requirements Document, the staff did not identify any issues that involve policy questions for the technical areas discussed in this chapter, other than those already identified in Commission papers listed in Appendix B to Chapter 1 of this report.

1.4 Outstanding Issues

The DSER for Chapter 3 of the passive Requirements Document contained the following outstanding issues:

Open Issues

P.3.0-1 leak testing of feedwater system valve that performs containment isolation function (5.5)

P.3.0-2 postaccident sampling system (PASS) (7)

Confirmatory Issues

P.3.C-1 anticipated-transient-without-scram (ATWS) events (5.5)

The final disposition of each outstanding issue is discussed in detail in the appropriate section of this chapter, as indicated by the parenthetical notation following the issue. All outstanding issues identified in the DSER for Chapter 3 have been resolved.

1.5 Vendor- or Utility-Specific Items

The DSER for Chapter 3 continued the following vendor- or utility-specific items. The final disposition of each item is discussed in the appropriate section of this chapter, as indicated by the parenthetical notation following each item.

P.3.V-1 vent and drain design (2.1.2)

P.3.V-2 reactor coolant interfacing systems design (2.1.7, 3.1, and 5.1)

P.3.V-3 specific performance requirements and acceptance criteria for active non-safety auxiliary systems (2.2, 6, and 8)

P.3.V-4 safety analyses of the abnormal conditions associated with loss of a feed pump or load rejection (3.2)

P.3.V-5 acceptability of operation with reactor coolant pump(s) (RCPs) out of service (3.2)

P.3.V-6 low-temperature overpressure protection system design details (3.3)

P.3.V-7 design analysis to confirm the capability and reliability of the passive decay heat removal system (PDHRs) (3.4)

P.3.V-8 specific design requirements for reactor vessel level instrumentation system (3.5)

P.3.V-9 design details of the man-machine interface system (M-MIS) for steam generator water level control (4.5)

P.3.V-10 design details to mitigate excessive leakage of main steam isolation valves (5.4)

P.3.V-11 design details for automatic reactor vessel overfill protection (5.5)

P.3.V-12 design details for PWR auxiliary systems, including chemical and volume control system (CVCS), against Standard Review Plan (SRP) Section 9.3.4 (6)

P.3.V-13 design details for reactor shutdown cooling pump seals (9)

P.3.V-14 adequate vendor assessment of shutdown and low-power operation risk (9)

The following vendor- or utility-specific items remain and will be addressed during the staff's review of an individual application for FDA/DC. The staff's position on each item is provided in the appropriate section of this chapter, as indicated by the parenthetical notation following each item.

P.3.V-1 vent and drain design (2.1.2)

P.3.V-2 reactor coolant interfacing systems design (2.1.7, 3.1, 5.1, and 9)

P.3.V-3 regulatory treatment of non-safety systems (2.2 and 6)

P.3.V-4 safety analyses of abnormal conditions associated with loss of feed pump or load rejection (3.2)

P.3.V-5 acceptability of operation with RCP(s) out of service (3.2)

P.3.V-6 low-temperature overpressure protection system design details (3.3)

P.3.V-7 design analysis to confirm the capability and reliability of the PDHRS (3.4)

P.3.V-8 design-basis analysis of steam generator tube rupture events (4.1)

P.3.V-9 design details of the M-MIS for steam generator water level control (4.5)

P.3.V-10 design details for automatic reactor vessel overfill protection (5.5)

P.3.V-11 effect of natural circulation in the analysis of ATWS events (5.5)

P.3.V-12 compliance of PASS with requirements of Item II.B.3 of NUREG-0737 (7)

P.3.V-13 design details for reactor shutdown cooling system (9)

P.3.V-14 requirements arising from resolution of issues identified in NUREG-1449 (9)

2 TOP LEVEL AND COMMON REQUIREMENTS FOR PWR AND BWR

2.1 Common Requirements

Section 2 of Chapter 3 of the passive Requirements Document contains those requirements established by the ALWR Utility Steering Committee for the reactor coolant system and non-safety-related auxiliary systems that are common to BWRs and PWRs. These requirements pertain to boundaries and interfaces with other plant systems, general system features, instrumentation and controls, and maintenance.

2.1.1 Protection of Components Inside Containment

Section 2.2.1.1 of Chapter 3 states that components that are inside the containment and required to function following a loss-of-coolant accident (LOCA) will be protected and located so that they will not be damaged by hydrogen burn or flooded as a result of a LOCA. The system arrangement and component design will provide protection against jets due to leaks in nearby high-pressure systems. Section 2.2.1.2 of Chapter 3 states that the plant designer will specify the environmental qualification requirements for critical components located inside the containment and required to function following a LOCA. The requirements to be specified include preparation and control of documentation packages demonstrating compliance with NRC requirements. The staff concludes that the above requirements do not conflict with any regulatory requirements. Environmental qualification of components is evaluated further in Section 4.8.2 of Chapter 1 of this report.

2.1.2 Draining, Filling, and Hydrostatic Testing

Section 2.2.4.1 of Chapter 3 requires that (1) vents, drains, and suitable isolation valves be provided for draining, filling, and venting the reactor coolant system (RCS) and non-safety-related auxiliary systems; (2) components and piping be arranged so that there are no undrained low points in the systems; and (3) vents be routed to a common collection system or to a common reference pressure point. However, no requirement is specified regarding a staff concern about the possibility of a LOCA outside the containment. In response to a staff question, EPRI stated in a letter dated May 13, 1991, that the BWR main steam drain and vent systems are required to meet the isolation valve requirements of General Design Criteria 54 and 55 of 10 CFR Part 50, Appendix A, and are to be designed for full RCS pressure through the outermost isolation valve. Piping beyond this point is to be designed for the expected pressure when blowing down these lines to the main condenser with the reactor at full pressure. The staff's position is that the vent and drain design of each ALWR should be consistent with the requirements for the reactor coolant interface systems in Section 2.2.14 of Chapter 3 of the passive Requirements Document. The staff will evaluate this issue further during its review of an individual application for FDA/DC.

2.1.3 Provisions for Decontamination

Section 2.2.5 of Chapter 3 specifies that the RCS design will include provisions to facilitate chemical decontamination to reduce shutdown radiation levels in piping and components. These provisions will include drain and

flush connections to ensure removal of decontamination fluids, in-line components to minimize trapping of decontamination fluids, and curvatures and connections to allow a probe (i.e., a hydro-laser) to pass inside the piping. The RCS will be designed to permit the use of mechanical decontamination devices. The plant designer should ensure that materials to be exposed to the decontamination fluids will be compatible with the reagents or chemicals to be used or that they will be designed to permit replacement. To facilitate the installation and operation of the equipment needed for full or partial decontamination of the RCS, staging areas are needed in the plant for this equipment. The passive Requirements Document (Chapter 3, Section 2.2.5.1; Chapter 6, Section 2.4.1; and Chapter 9, Sections 6.1.1 and 6.1.3) states that ALWRs will be designed to (1) include provisions such as staging areas to facilitate chemical decontamination and (2) interface with major decontamination equipment. The staff concludes that the above requirements will result in lower personnel exposures while the system is being decontaminated and, therefore, are acceptable.

2.1.4 Insulation

Section 2.2.7 of Chapter 3 requires that designated piping and components be provided with insulation that can be removed quickly and can be reused. This insulation will be lightweight to facilitate quick removal and installation in those areas where external access is required for inservice inspection. Adequate laydown storage space will be provided for removed insulation. The staff concludes that this requirement is acceptable because these features will reduce the time required to perform inservice inspections on piping and components in high-radiation areas and, therefore, will result in dose savings.

Section 2.2.7.4 of Chapter 3 requires that material used in the insulation system contain a limited amount of corrosive chemicals so as not to contribute to stress corrosion cracking of austenitic stainless steel surfaces. The staff concludes that this requirement is acceptable because it is consistent with the guidelines in Regulatory Guide (RG) 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

2.1.5 Corrosion/Erosion Potential

Section 2.2.9 of Chapter 3 requires that the construction materials, sizing, and geometries of piping that will carry high-energy fluids (e.g., feedwater, wet steam) be such as to reduce the potential for corrosion/erosion of the piping. Piping areas that might be subject to corrosion/erosion should be capable of being inspected. The staff agrees that these requirements will reasonably minimize the potential for corrosion/erosion of piping and, therefore, finds them acceptable.

2.1.6 Utilization of Robotics

Section 2.2.13 of Chapter 3 requires that the RCS arrangement facilitate the use of robotic technology for inspection and maintenance operations. Nuts on all major openings of the reactor coolant pressure boundary will be designed to facilitate engagement by a remote handling or robotics tool. The staff

agrees that the use of robotics for inspection and maintenance in radiation areas will reduce personnel doses and, therefore, finds this requirement acceptable.

2.1.7 Reactor Coolant Interface Systems

In Chapter 3, Revision 0, of the passive Requirements Document, EPRI did not specify requirements regarding protection against a LOCA outside the containment for those systems interfacing with the RCS. The staff stated its position in a request for additional information dated February 28, 1991, regarding an interfacing system LOCA (ISLOCA): future ALWR designs are required to reduce the possibility of a LOCA outside the containment by designing, to the extent practicable, all systems and subsystems connected to the RCS to an ultimate rupture strength (URS) at least equal to full RCS pressure. The staff included the phrase "extent practicable" in recognition that all systems must eventually interface with atmospheric pressure and that for certain large tanks and heat exchangers, it would be difficult or prohibitively expensive to design such systems to the URS equal to full RCS pressure. For those interfacing systems and components that cannot meet the RCS full-pressure URS criterion, the staff also will require that the plant designer provide (1) justification based on engineering feasibility analysis as to why it is not practicable to use components that have been designed to meet the RCS full-pressure URS criterion and (2) compensating isolation capability and protective measures specified in the Commission paper SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs," issued on April 2, 1993.

In response to the staff question, EPRI revised Chapter 3 in Revision 2 by adding the requirements of Section 2.2.14. These requirements include the following:

- All systems and subsystems connected to the RCS that will extend outside the primary containment boundary will be designed, to the extent practicable, to an ultimate URS at least equal to full RCS pressure.

- For those interfacing systems or subsystems that do not meet the RCS full-pressure URS criterion, the plant designer will have to determine by evaluation if the degree and quality of isolation or reduced severity of the potential pressure challenges is low enough to preclude an ISLOCA.

- Each of the high- to low-pressure interfaces (i.e., between those systems normally exposed to the RCS or RCS pressure and the interfacing lower pressure system) will include the following protection measures: (1) the capability for leak testing of the pressure isolation valves, (2) valve position indication that is available in the main control room even when isolation valve operators are deenergized, and (3) high-pressure alarms to warn control room operators when rising RCS pressure approaches the design pressure of the attached low-pressure systems and both isolation valves are not closed.

These requirements are, in general, consistent with the NRC staff position. However, it will be necessary for the plant designer to demonstrate during the

NRC staff's review of an individual application for FDA/DC that any RCS interfacing system whose URS is not at least equal to full RCS pressure could not be practically designed to meet such a criterion.

It should be noted that the degree of isolation or number of barriers (e.g., three isolation valves) is not sufficient justification for using low-pressure components that can be practically designed to the full RCS URS criterion. It is the staff's position that piping runs should be designed to meet the RCS full-pressure URS criterion, as should all associated elements such as flanges, connectors, packing, valve stem seals, pump seals, heat exchanger tubes, valve bonnets, and RCS drain and vent lines. The plant designer should make every effort to reduce the level of pressure challenge to all systems and subsystems connected to the RCS.

For all interfacing systems and components that do not meet the RCS full-pressure URS criterion, the staff requires that the plant designer justify why it is not practicable to reduce the pressure challenge any further. This justification must be based on engineering feasibility analysis and not solely on risk benefit tradeoffs.

For those interfacing systems and components for which the impracticability of full RCS pressure capability has been acceptably justified, the plant designer must demonstrate compensating isolation capability. For example, the designer should demonstrate for each interfacing system and component that the degree and quality of isolation or reduced severity of the potential pressure challenges compensates for and justifies the safety of the low-pressure interfacing system or component. Adequacy of pressure relief and piping of relief back to the primary containment are possible considerations. The staff will address these issues during its review of an individual application for FDA/DC.

2.2 Non-Safety Auxiliary Systems

The passive Requirements Document proposes passive ALWR designs that require all safety systems that supply safety injection water and provide core and containment cooling to be of passive designs. All active auxiliary systems are designated "non-safety systems."

Section 2.3.1 of Chapter 3 specifies the non-safety auxiliary systems that will provide defense-in-depth capability for reactor coolant makeup and decay heat removal. These systems include (1) the chemical and volume control system (CVCS) and control rod drive (CRD) system for reactor coolant makeup for PWRs and BWRs, respectively; (2) the reactor shutdown cooling (RSDC) system and backup feedwater pumps, and the reactor water cleanup (RWCU) system for PWR and BWR decay heat removal, respectively; (3) the fuel pool cooling and cleanup system (FPCCS) for spent fuel decay heat removal; and (4) those systems and structures needed to support these functions. These systems will not be designed to meet safety-grade criteria, and credit will not be taken for them in the licensing design-basis-accident analyses. Many of these systems are traditionally safety-grade systems in the current LWR designs.

However, the passive Requirements Document requires that these active systems provide defense-in-depth capabilities by serving as the first line of defense in the event of transients and plant upsets to reduce challenges to the

passive safety systems. For example, Section 2.3.2 of Chapter 3 requires at least one non-safety auxiliary system to provide means, without reliance on any safety system except the reactor protection system, to maintain reactor coolant inventory and remove decay heat for the plant during frequent and moderate-frequency events to ensure compliance with the specified design limits; to serve as the first line of defense for pipe breaks having a break area equivalent to that of a pipe with an inside diameter of 3/8 inch; and to ensure that operation of the safety depressurization system is a very low probability event.

In its letter dated May 26, 1993, EPRI proposed an approach to determine which system should receive which kind of regulatory oversight. A key element of this process is to ensure that the passive plant vendors use probabilistic risk assessment methods to examine the reliability of the safety and non-safety systems and ensure that important risk insights are considered in the design. This process, and the staff's evaluation of it, are described in greater detail in SECY-94-084 in Annex B of Appendix B to Chapter 1 of this report. In its SRM dated June 30, 1994, the Commission approved the staff's recommendation regarding the regulatory treatment of non-safety systems (RTNSS). The staff will review the implementation of EPRI's process during its review of an individual application for FDA/DC to ensure that passive plant vendors follow this process satisfactorily.

The passive safety systems are designed to use natural forces (such as buoyancy, gravity, and stored energy) to supply safety injection water and provide core and containment cooling. These passive systems do not include pumps and include only those valves that operate by dc electric power (by means of batteries) or air or that are check valves operating by means of the pressure differential across the valve. The staff concludes that uncertainties are associated with these unique passive features because there is a lack of proven operational performance history. For example, there are some uncertainties in overall functional performance of the passive core and containment heat removal. There are also uncertainties regarding the operability of check valves in the passive safety systems at low differential pressures during natural circulation or gravity injection. These low driving forces may provide insufficient means of forcing operation of stuck check valves, unlike the present active emergency core cooling systems, which can overcome such stuck valves. Comprehensive test programs now under way for the specific designs and analytical assessments are expected to address these issues and provide an adequate basis for supporting passive safety system functional performance, yet possible uncertainties may remain that emphasize the importance of the active non-safety systems in providing the defense-in-depth for preventing and mitigating accidents and core damage. Therefore, in its review of the passive designs, the staff will evaluate not only the passive safety systems, but also the functional capability and availability of the active non-safety systems in accordance with EPRI's RTNSS process described above to ensure a robust defense-in-depth and the capability for preventing accidents and core damage.

The passive Requirements Document does specify the system and equipment design requirements for the defense-in-depth active systems. These requirements

include radiation shielding, redundancy and single active failure considerations, electric power availability, protection against internal flooding and in-plant hazards, and testing to demonstrate system capability. For example, Section 2.3.3 of Chapter 3 requires the non-safety auxiliary system to

- be arranged and shielded to permit access for operation and maintenance under conditions anticipated during and after such events and to meet the requirements for equipment survivability (Section 6.6.5.4, Chapter 5) if identified for use as part of a severe accident management program (Note: This requirement was added to Section 2.3.3.1 of Chapter 3 in Revision 2 in response to a staff question)

- be redundant to ensure the defense-in-depth of each system, assuming either (1) a single active failure of equipment that must change state or position to perform the defense-in-depth function or (2) unavailability of this equipment because of maintenance

- have available electric power from normal station ac and onsite non-safety ac power supplies; those power supplies to the redundant equipment (or trains) are to be separated to the extent practicable (i.e., power from separate buses)

- have redundant components that are protected against internal flooding and other in-plant hazards (Section 2.3.6, Chapter 6)

- be designed to include demonstration by test of system capability to satisfy the defense-in-depth requirements in system designs analyses.

Section 2.4.2.8 of Chapter 5 also requires that the non-safety-related equipment necessary for plant recovery after the assumed 72-hour accident duration be designed for the expected environment during the 72-hour period and during its operation after 72 hours.

These objectives are noteworthy and provide important high-level goals for non-safety systems. However, during an FDA/DC review, the staff will evaluate these issues in accordance with EPRI's RTNSS process to ensure high quality of the non-safety systems. For example, the scope and degree of required redundancy and separation will have to be ensured, as will the methodology that is used to identify the limiting single active failure. The staff will also assess the degree of fire protection provided and the initiating events and plant upsets considered by the designer to demonstrate that the non-safety systems meet their defense-in-depth roles. The FDA/DC applicant should also indicate which design criteria (e.g., Institute of Electrical and Electronics Engineers 279) for electrical system quality and separation (or lesser subsets) are used to meet the EPRI goals of ensuring high quality of the non-safety systems.

The passive Requirements Document requires at least one non-safety system to ensure compliance in meeting the EPRI ALWR safety goals, such as providing decay heat removal function given a moderate-frequency event. The staff agrees that this is an important goal. It also believes that the active systems' capabilities and reliabilities should be influenced by their safety

importance and that each plant designer should demonstrate that the capabilities and importance of the non-safety systems have been considered during the design process.

The staff concludes that the passive Requirements Document provides important high-level goals for non-safety systems. However, since design detail is not yet available, the staff expects each designer to demonstrate a high level of assurance that these systems will be available when needed and that they will respond properly when challenged.

In a position paper dated March 19, 1992, EPRI stated that active non-safety systems in the passive designs are provided for utility investment protection, not for plant safety or meeting regulatory requirements, and that passive safety systems and additional safety features (i.e., non-safety-grade systems required by regulation to address specific accident considerations, such as ATWS, station blackout) alone meet the Commission's regulations and safety goals. However, the staff believes that, because of limited operational experience and low driving force of the passive safety systems, uncertainties exist concerning the performance of the passive safety systems and overall performance of the reactor coolant makeup and core and containment heat removal, and therefore, regulatory oversights of certain non-safety systems may be required depending on their risk significance. These active non-safety systems will not be required to meet all the safety-related criteria, but there should be a high level of confidence that those active systems that have a significant safety role are designed in accordance with their performance and reliability missions to ensure their availability when needed.

The staff and EPRI have had many interactions to resolve the issue of RTNSS and have agreed on a structured process to be used by plant designers for evaluating the RTNSS issue. This RTNSS process, described in an EPRI letter of May 26, 1993, involves use of a "focused PRA" to determine any combination of systems, structures, and components (SSC) necessary to meet NRC regulations, safety goal guidelines, and containment performance goal objectives, and evaluate the risk significance and reliability/availability (R/A) missions of the SSCs. In principle, all non-safety SSCs in the focused PRA model needed to meet NRC requirements, safety goal guidelines, and containment performance goals are potentially subject to proper regulatory oversight. On the basis of their risk significance and R/A missions, the plant designer will also propose for staff consideration appropriate regulatory oversight measures for these non-safety SSCs. These measures may include a proper operational reliability assurance program and maintenance rule implementation, and proper short-term availability control mechanisms such as technical specification limiting conditions for operation and administrative control of shutdown configuration. In addition, deterministic requirements for risk-significant SSCs will be included in the design certification rule. These regulatory oversights will ensure that the risk-significant non-safety SSCs are designed, constructed, maintained, and operated in accordance with their R/A missions.

The construction of the focused PRA is based on a comprehensive design-specific Level 3 PRA that includes all appropriate internal and external events and power and shutdown operations. The scope of the RTNSS evaluation includes all SSC functions that are relied on to meet the beyond-design-basis deterministic NRC performance requirements, to resolve long-term safety

(beyond 72 hours), to address seismic events, to meet the containment performance goal and containment bypass concerns, and to prevent significant adverse systems interaction, under power and shutdown operations. Therefore, the process provides a comprehensive approach that will address the RTNSS and other relevant issues as integral parts of the process. The R/A missions and the proper regulatory oversight of the risk-significant non-safety systems determined as a result of this evaluation will properly address the concerns of many issues, such as the stable safe shutdown requirements and related passive system design basis of 72-hour capability, station blackout, electrical distribution, and control room habitability.

The staff concludes that the RTNSS process described in EPRI's May 26, 1993, letter is acceptable for resolving the RTNSS issue. In Revision 5, EPRI provided the requirements pertaining to the RTNSS process in Section 2.5.1.4 of Chapter 1 and Section 1.6 of Appendix A to Chapter 1. This RTNSS process is required to be integrated into the design process. Because the design requirements for non-safety systems are dependent on the R/A missions of the risk-significant non-safety systems, no conclusion can be reached regarding risk-significance of non-safety systems until an appropriate determination of R/A missions using a design-specific focused PRA is completed. Plant designers are required to use the results from such a study to identify risk-significant important systems and their R/A missions and provide comparisons with the safety goal guidelines, and report this information in the PRA. The staff will review the implementation of this RTNSS process during its review of an individual application for FDA/DC to ensure that passive plant vendors follow this process satisfactorily.

The RTNSS process is described in greater detail in SECY-94-084 in Annex B of Appendix B to Chapter 1 of this report.

3 PWR REACTOR COOLANT SYSTEM

The PWR RCS and its boundaries will include the pressurizer, the pressurizer safety valves, the reactor coolant pumps, the reactor coolant main loop piping up to the welds connecting the reactor vessel, the reactor coolant connections to the steam generators, and the steam generator tubes. The RCS will transfer heat energy produced in the reactor core to the steam generators by forced circulation of hot reactor coolant from the reactor vessel to the steam generators and return the cooled reactor coolant from the steam generators to the reactor vessel. During both normal and abnormal operating conditions, pressure in the RCS is required to be maintained by the pressurizer within specified limits without actuation of pressure relief devices. The RCS is also required to (1) enable natural circulation for decay heat removal through the steam generator, (2) provide RCS overpressure protection by pressure relief devices for all design-basis events, (3) maintain a single major steam-water interface in the pressurizer where it can be monitored for all normal and abnormal operating conditions, and (4) provide high point venting of non-condensible gases in the RCS following severe accidents.

3.1 PWR Interfacing System Loss-of-Coolant Accident

Section 3.1.3 of Chapter 3 of the passive Requirements Document identifies the systems that will interface with the PWR RCS. In response to a staff question dated May 13, 1991, regarding interfacing system LOCA (ISLOCA), EPRI submitted its evaluation of the following interfacing systems: chemical and volume control system (CVCS), passive decay heat removal system (PDHRS), depressurization system, passive safety injection system (PSIS), reactor shutdown cooling (RSDC) system, and process sampling system. EPRI concluded that the design of these systems and subsystems interfacing with the PWR RCS will be in full compliance with the requirements for ISLOCA protection as specified in Section 2.2.14 of Chapter 3. However, certain portions of these systems, as described below, do not meet the RCS full-pressure ultimate rupture strength (URS) criterion:

- CVCS components upstream of charging pumps (e.g., boric acid storage tank, as well as discharge lines to the waste processing system) are designed for process conditions.

- Portions of the PDHRS and PSIS drains beyond the outboard isolation valves to the liquid radioactive waste processing system are designed for process conditions.

- Vents and drain systems of the RSDC system beyond the isolation valves are designed for process conditions.

EPRI contends that these systems are beyond the "extent practicable" definition of Section 2.2.14.1 of Chapter 3, but does not provide bases to justify the impracticality of designing them to meet the RCS full-pressure URS criterion.

As stated in Section 2.1.7 of Chapter 3 of this report, the staff's position is that, to the extent practicable, piping runs and associated elements should be designed to meet the RCS full-pressure URS criterion, and that the degree

of isolation is not sufficient justification for using low-pressure components. For all interfacing systems and components that do not meet the RCS full-pressure URS criterion, the staff requires that the plant designer justify the impracticality of designing a low-pressure system to the URS criterion to reduce the pressure challenge. While reviewing an individual application for FDA/DC, the staff will further evaluate the design of the interfacing systems, including justifications for impracticality if a system component is not designed to meet the RCS full-pressure URS criterion.

3.2 RCS Manual Maneuvering Capability

Section 3.2.1.2 of Chapter 3 of the passive Requirements Document specifies that automatic override action in the manual reactivity control mode to reduce power may be used to avoid a reactor trip following a loss of a main feed pump or load rejection. This is acceptable provided the plant designer performs appropriate safety analyses to determine the reduced power levels for the abnormal conditions and to demonstrate that the specified acceptable design limits are not exceeded for these conditions. The staff will evaluate the analyses during its review of an individual application for FDA/DC.

Section 3.2.4 of Chapter 3 originally required the RCS to be designed to allow operation with "at least one" reactor coolant pump (RCP) out of service and indicated that the plant designer will have to perform a plant-specific analysis to determine the power level at which operation with one RCP out of service is permitted. In its letter dated May 22, 1991, EPRI indicated that it did not intend to preclude operation with more than one pump out of service, and if the plant designer claims that more than one pump out of service can be allowed, the designer will have to perform an analysis to support that claim. However, in Revision 3, EPRI deleted Section 3.2.4 from the passive Requirements Document. Therefore, there is no longer a requirement for a system capability to allow operation with an RCP out of service. The staff concludes that deletion of Section 3.2.4 is acceptable and resolves the staff's concern in this area by eliminating the offensive requirement.

The issue of one out-of-service pump is also addressed in Section 3.5 of Chapter 1 of this report. During its review of an individual application for FDA/DC, the staff will evaluate the concerns delineated in Generic Letter (GL) 86-09, "Technical Resolution of Generic Issue No. B-59-(N-1) Loop Operation in BWRs and PWRs." In addition, if more than one RCP is allowed to be out of service, the plant designer should establish limitations, such as system configuration, on allowed out-of-service pumps on the basis of a plant-specific analysis demonstrating compliance with acceptance criteria.

3.3 Low-Temperature Overpressure Protection

In Branch Technical Position RSB 5-2 of SRP Section 5.2.2, "Overpressure Protection," the staff specifies that a system be designed and installed that will prevent the facility from exceeding the limits in applicable technical specifications and Appendix G of 10 CFR Part 50 for the RCS while operating at low temperature, and that the low-temperature overpressure protection (LTOP) system be operable during startup and shutdown conditions below the "enable temperature," which is defined, in RSB 5-2, as the water temperature corresponding to a metal temperature of at least the nil ductility transition

temperature plus 90 °F at the belt line location. The LTOP system is required to be able to perform its function assuming any single active component failure.

Section 3.3.2 of Chapter 3 of the passive Requirements Document requires that (1) the RCS LTOP be provided by devices (relief valves) in the reactor shutdown cooling (RSDC) system with the relief capacity for LTOP to be sized (as required by Section 9.3.3 of Chapter 3) with the minimum calculated end-of-life pressure relief setpoints; (2) the setpoint of the devices that provide overpressure protection for normal operation not be reset to provide overpressure protection at low temperature; and (3) the nil ductility transition temperature of ferritic reactor coolant pressure boundary materials not exceed 10 °F.

Generic Safety Issue (GSI) 94, "Additional LTOP for LWRs," which addresses the staff's concern regarding availability of overpressure protection when the reactor is water-solid, is discussed in Section 3.3.2 of Appendix B to Chapter 1 of the passive Requirements Document. EPRI states that the ALWR design will not require any special LTOP since material requirements have been established to ensure an acceptably low nil ductility transition temperature at the end-of-life conditions, and relief capacity for LTOP will be provided for most limiting events. In a letter dated May 13, 1991, EPRI also indicated that GL 90-06, "Additional Low-Temperature Overpressure Protection for Light Water Reactors," does not apply to ALWRs because there are no power-operated relief valves and block valves in the ALWR designs.

However, the staff position stated in Enclosure B of GL 90-06 regarding GSI 94 applies to all PWRs whether or not they have power-operated relief valves and block valves. Therefore, the GL 90-06 guidance to ensure availability of LTOP protection by limiting the allowable outage time to 24 hours for a single LTOP protection channel while operating in Modes 5 or 6 is applicable to the passive PWR plants using the reactor shutdown cooling system safety relief valves for LTOP protection. The staff also notes that the GL 90-06 guidance applies to operational control and should be addressed on a design-specific basis.

Since the reactor shutdown cooling system is not required to be designed to satisfy safety-related requirements, it is necessary to further evaluate the appropriateness of relying on the relief valves of the RSDC system for RCS overpressure protection at low-temperature conditions. This is part of the overall concern regarding the regulatory treatment of non-safety systems in the passive plants discussed in Section 2.3.1 of Chapter 1 of this report (also discussed under Item III of Annex A of Appendix B to Chapter 1 of this report). As discussed in Section 2.2 of this chapter, EPRI has developed an acceptable process for resolving the RTNSS issue. The process will be used by the plant designers to determine any combination of systems, structures, and components needed to meet the NRC deterministic requirements, safety goal guidelines, and containment performance goals. Non-safety systems, including the LTOP systems, will be an integral part of the RTNSS evaluation process, and any requirements, including those specified in GL 90-06, pertaining to the LTOP systems will be determined as a result of analysis performed by the plant

designers based on their risk-significance. The staff will evaluate the design of the LTOP systems and associated requirements, such as GL 90-06, during its review of an individual application for FDA/DC.

Reactor Coolant Water Chemistry

Section 3.3.3 of Chapter 3 requires that the plant design provide the system features that will facilitate compliance with EPRI Report NP-7077, "Primary Water Chemistry Guidelines," Revision 2. In addition to the limitations on reactor coolant makeup in EPRI Report NP-7077, the makeup in the RCS is to meet the parameter limits in Section 5 of Chapter 1 of the passive Requirements Document.

The staff concludes that these requirements are acceptable. Water chemistry limitations are included in Section 5 of Chapter 1, which the staff found acceptable.

Leak-Detection Capability

Section 3.3.4 of Chapter 3 states that the RCS design will provide for the detection of leakage from the RCS so that pressure boundary leakage can be detected with adequate confidence to support the leak-before-break methodology. Section 7.14 of Chapter 10 states that the man-machine interface systems for reactor coolant leak detection will provide the monitoring necessary to ensure that RCS leakage is within acceptable limits. In Appendix B to Chapter 1, EPRI committed to comply with the staff's review guidance in SRP Section 5.2.5, "Reactor Coolant Pressure Boundary Leakage Detection." The staff concludes that these EPRI requirements and commitment are acceptable.

3.4 Reactor Coolant Piping and Connections

Section 3.4.1 of Chapter 3 of the passive Requirements Document requires that the reactor coolant piping and connections to the RCS meet the material requirements in Section 5 of Chapter 1 and that all materials be identified.

The following requirements are also specified:

- Reactor coolant piping will be seamless (longitudinal welds are not permitted).

- Forging will be the reference fabrication method used for the reactor coolant piping. Cast pipe material may be used only if a volumetric inspection technique has been qualified and it can be established that the cast material will not result in more inservice inspection than would be required for the preferred forged pipe.

The staff concludes that these requirements are acceptable. Forged piping requires less inservice inspection; therefore, the use of forged and seamless reactor coolant piping in the ALWR will result in lower overall personnel radiation doses.

Reactor Coolant Pumps

Section 3.4.2.1 of Chapter 3 requires that the reactor coolant pumps (RCPs) be categorized as critical components as defined in Section 5 of Chapter 1 of the passive Requirement Document. All materials used are to be identified and are to meet the requirements of Section 5 of Chapter 1.

All RCP parts wetted by primary coolant will be compatible with the solutions used in the reference decontamination method. If they are not compatible, a method will be defined, as part of the plant design, to ensure that incompatible parts can be removed from the pump or isolated from the decontamination solution (e.g., by providing a purge flow) without a loss of an effective decontamination. It is acceptable to remove and replace incompatible parts after decontamination, provided leaving a part installed during decontamination does not reduce the effectiveness of the decontamination or affect other plant components. The plant designer is required to identify any parts incompatible with the decontamination solution, as well as any incompatible parts that are not protected by a purge flow and must be removed from the pump before or after decontamination.

The staff concludes that these requirements are acceptable because pump materials will be compatible with the primary coolant and will be identified as critical components by the plant designer.

Because an RCP flywheel has a large mass and typically rotates at speeds of 900 to 1200 rpm during normal operation, a loss of flywheel integrity could result in high-energy missiles and excessive vibration of the RCP assembly.

Section 3.4.2.10 of Chapter 3 requires that if an internal flywheel is used in the RCP, the plant designer must demonstrate that, in the event of a flywheel failure, the surrounding pump components are adequate to prevent missiles from jeopardizing the structural integrity of the reactor coolant system and other plant structures, systems, and components.

If an internal flywheel is used in the RCP, the design and construction of the flywheel must be such that areas of high stress concentration are minimized, suitable material toughness is provided, and all the fabrication and examination procedures necessary to achieve a high level of integrity throughout the plant operating life are implemented. The plant designer must show that the flywheel design is such that it does not add to the burden of pump maintenance and does not require periodic inservice inspection.

The staff concludes that these requirements are acceptable because they are consistent with the guidance in SRP Section 5.4.1.1, "Pump Flywheel Integrity."

Pressurizer

Section 3.4.3.1 of Chapter 3 defines the pressurizer as a critical component and requires that all materials be identified and meet the requirements of Section 5 of Chapter 1 of the passive Requirements Document. The staff reviewed and found acceptable the materials selection criteria in Section 5 of Chapter 1.

Sections 3.4.3.2.3 and 3.4.3.2.4 of Chapter 3 specify the requirements for pressurizer heater capacity and power sources for hot standby conditions. These sections require (1) the combination of maximum pressurizer heat loss and heater capacity to be such as to maintain the pressurizer at normal operating pressure during hot standby conditions and (2) the heater capacity to be provided by redundant groups of heaters, each of which will be capable of being powered from either offsite power or different onsite non-safety ac power sources. The staff concludes that the above requirements do not conflict with regulatory guidance.

However, the onsite non-safety ac power source requirement is inconsistent with the requirement of 10 CFR 50.34(f)(2)(xiii) to provide pressurizer heater power supply and associated motive and control power interfaces sufficient to establish and maintain natural circulation in hot standby conditions with only onsite power available. In Section 4.1 of Appendix B to Chapter 1 of the passive Requirements Document, EPRI states that this issue is in compliance with the intent of the regulation. EPRI contends that the intent of the regulation is to ensure availability of the primary safety-grade decay heat removal method, and that this intent is met by the passive decay heat removal system (PDHRS) and is not dependent on pressurizer heater control. In its letter dated May 13, 1991, regarding an assessment of the capability and reliability of the PDHRS under various transient and degraded plant conditions, EPRI indicated that the plant designer will perform the analysis during detailed design to confirm the capability and reliability of the PDHRS under various transient and degraded plant conditions (e.g., loss of all decay heat removal capability by the steam generators). This analysis will include an evaluation of the capability of the PDHRS to maintain hot standby conditions without pressurizer heater control of primary coolant pressure. Therefore, the staff will assess the adequacy of this approach for meeting 10 CFR 50.34(f)(2)(xiii) during its review of an individual application for FDA/DC. The plant designer will have to document any deviations from the applicable requirements in NUREG-0737, "Clarification of TMI Action Plan Requirements," at that time.

3.5 Postaccident Monitoring Instrumentation and Control

Section 3.5.2.1 of Chapter 3 of the passive Requirements Document requires that instrumentation to monitor adequate core cooling and the conditions following an accident be provided and states that the ALWR is intended to be consistent with the applicable regulatory guidance for monitoring postaccident conditions. However, in Section 3.5.2 of the original Requirements Document, EPRI did not list the reactor vessel level instrumentation system (RVLIS) as a part of monitoring capability. In Section 2.4.1 of Appendix B to Chapter 1, EPRI classified the elimination of the RVLIS as a plant optimization issue and gave the bases for eliminating this system. In Revision 3 of the passive Requirement Document, which was issued after the staff's DSER on passive plant designs, EPRI deleted Section 2.4.1.

As a resolution of Three Mile Island (TMI) Action Plan Item II.F.2 (NUREG-0737), 10 CFR 50.34(f)(2)(xviii) requires commercial nuclear power plants to provide control room instrumentation to give "unambiguous indication of inadequate core cooling such as primary coolant saturation meters in PWRs, and a suitable combination of signals from indicators of coolant level in the reactor vessel and in-core thermocouples in PWRs and BWRs."

In evaluating proposed systems for operating plants and those seeking operating licenses, the staff has determined that vessel level instrumentation is a necessary part of the enhancements required to reduce the likelihood of approaching and exceeding adequate core cooling conditions.

The staff reviewed EPRI's bases and justification in Appendix B to Chapter 1 for not including a requirement for an RVLIS for the ALWR. In the DSER, the staff determined that EPRI did not present a sufficiently new argument to warrant altering the NRC position requiring the RVLIS for existing PWRs. The staff also did not find the design of the passive PWRs sufficiently different from current designs to support deviation from NRC's requirement for an RVLIS.

The larger pressurizer and dedicated vent system proposed for the passive plant will reduce the probability of loss of pressurizer level and will increase the plant's reliability. However, since these improvements do not eliminate the potential for a loss-of-pressurizer-level condition and, therefore, do not justify removal of an RVLIS, the staff concluded in the DSER that the passive Requirements Document does not meet the NRC requirement to have a RVLIS.

It was the objective of the policies established by NUREG-0737 and RG 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Condition During and Following an Accident," and the industry's development effort that reliable and unambiguous indication of inadequate core cooling (ICC) be provided. Although the systems designed in response to the need for clear and reliable indication and to these guidance documents are not ideal, they have been judged to satisfy the objectives to a satisfactory degree. EPRI's ALWR design criteria should require a system at least equal to those currently approved for current PWRs; however, to reduce concerns expressed by EPRI about the reliability and human factors aspect of the current-generation RVLIS, in the DSER, the staff recommended that EPRI consider requiring better system performance in its passive Requirements Document. Under most conditions, the RVLIS does provide the clear indication sought. The staff has concluded that the total ICC instrumentation, when supplemented by the subcooling margin monitor and core exit thermocouples, meets the established requirements. The same cannot be said for reliance on pressurizer level indication. As discussed in Section 2.4.1 of Appendix B to Chapter 1 of the original passive Requirements Document, the pressurizer level increases for breaks in some locations and decreases for breaks in other locations. The positive indication provided by the RVLIS, rather than diverting the operator's attention, may provide the information for the operator to make a proper decision.

The RVLIS has already been shown to be a useful instrument in operating reactors. For example, at North Anna Unit 1, between June 17 and June 21, 1987, approximately 17,000 gallons of RCS inventory was removed from the RCS. The pathway for the loss was through the pump backseat of the "A" reactor coolant pump, up the shaft, and out through the seals into the containment sump. The pump had been decoupled from the motor for stator replacement. The inventory loss was not detected because the pressurizer had been allowed to cool down, a vacuum had developed in the pressurizer, and a bubble had formed in the reactor vessel head. In this condition, pressurizer level indication did not adequately represent RCS inventory. Indication was available through the RVLIS portion of the integrated core cooling monitoring system. Even

though the RVLIS had not been declared operational and operators did not act to terminate the condition on the basis of that information, it did alert the operators, and action was finally taken when the vacuum in the pressurizer was also detected. At that time, the RVLIS was used to track the vessel level during venting of the reactor head and refilling of the system.

The staff has also found other events during which the RVLIS (if it had been installed) could have resulted in better operator response. These included a natural circulation cooldown event at St. Lucie Unit 1 during which a bubble formed in the reactor vessel head, even though there was continued indication of adequate subcooling margin.

This instrumentation should meet the staff requirements for vessel level instrumentation as part of an inadequate core cooling detection system and yields very useful information to be considered during accident management activities. Although accident management guidelines are being developed by vendor owners groups, the staff expects that indication of vessel level will be an important parameter for preventing core damage or possible arresting core-melt progression before vessel failure. Therefore, in the DSER, the staff concluded that each ALWR design must have an RVLIS and that each PWR ALWR designer must identify system design and performance criteria, including the system's potential accident management role and the resulting severe environment it may be subjected to. In Revision 4 of the passive Requirements Document, EPRI revised Section 6.3.3.2 of Chapter 4 to require an RVLIS in passive PWRs. The staff concludes that this is acceptable.

Loose Parts Monitoring

Section 3.5.2.2 of Chapter 3 requires instrumentation to be provided to detect the presence of loose parts in the reactor coolant system. Table B.1-2 of Appendix B to Chapter 1 indicates that EPRI commits to comply with the guidelines in RG 1.133, Revision 1, "Loose-Part Detection Program for Primary System of Light-Water-Cooled Reactors." With this commitment, the requirement is acceptable.

4 STEAM GENERATOR SYSTEM (PWR)

Section 4.1.2 of Chapter 3 of the passive Requirements Document specifies the functional requirements for PWR steam generators: (1) produce steam with no more than 0.25-percent moisture carryover, (2) provide the capability for continuous hot blowdown of the secondary side, (3) provide indication and control of secondary-side water level, (4) provide a leaktight boundary between the primary and secondary sides, (5) serve as the primary means for removing decay heat from the reactor coolant during plant shutdown, and (6) provide full wet layup of the steam generator under deoxygenated, pH-controlled conditions.

4.1 Steam Generator Tube Rupture

EPRI revised Section 4.2.5 of Chapter 3 for Revision 4 of the passive Requirements Document. The coping capability of the reactor coolant system (RCS) design for the rupture of a single steam generator tube (single SGTR) is specified depending on whether there is operator action. Section 4.2.5.1.1 requires the plant to be designed so that, with operator actions within 72 hours but no sooner than 30 minutes, the combined capability of plant trip, turbine bypass system actuation, and controlled depressurization of the RCS will prevent the actuation of steam-side safety valves in the event of a complete and sudden rupture of one steam generator tube. Section 4.2.5.1.2 requires the plant designer to ensure that the plant system designs provide reliable and rapid means for the operators to detect a primary-to-secondary leak up to an SGTR and determine the affected steam generator.

Section 4.2.5.1.3 requires that, without operator action, means be provided to automatically equalize primary and secondary-side pressures following a single SGTR, without water relief through the secondary-side safety valves or power-operated relief valves and without initiating the automatic depressurization system. Although the passive Requirements Document also requires that these coping capabilities be demonstrated on a best-estimate (BE) basis taking credit for non-safety systems, such as control-grade turbine bypass system, the staff finds these objectives and required measures to avoid lifting of steam-side safety valve or water relief by means of steam-side safety valves for an SGTR to be acceptable since the radioactive release outside the containment will be minimized. However, in reviewing an application for FDA/DC, the staff will require a design-basis analysis of single SGTR events taking credit for only the safety systems to demonstrate that the radiological consequence is within the acceptance limit. Section 2.3.2 of Chapter 1 of the passive Requirements Document specifies the licensing design-basis requirements, and steam generator tube rupture is included in Table 1.2-1 as one of the initiating events to be analyzed. This is, therefore, acceptable.

In Revision 4 of the passive Requirements Document, EPRI added Section 4.2.5.2 to Chapter 3 specifying that, on a BE basis, the designer must analyze the plant response for a spectrum of multiple SGTRs to demonstrate that sufficient success paths are included in the design so that multiple SGTR scenerios are not significant contributors to core damage or release frequencies. EPRI added this requirement to address a staff concern stated in a draft Commission paper dated June 25, 1992, that the passive ALWR plant may respond to a multiple SGTR event in a manner considerably different from that for a single

SGTR, and the consequences may also differ significantly. EPRI maintained that the licensing design-basis SGTR should remain a single tube rupture. The staff determined that the licensing analysis of SGTR for passive PWRs may need to be extended from a single SGTR to multiple tube ruptures because it estimated that the frequency of occurrence of multiple SGTR events could be the same order of magnitude as that other design-basis accidents, such as large- and small-break loss-of-coolant accidents. In SECY-93-087, the staff recommended that analysis of multiple SGTRs involving two to five steam generator tubes be included in an application for FDA/DC for passive PWRs. The staff also stated that it will evaluate these analyses during the FDA/DC review process to help determine the number of SGTRs that will be incorporated into the PWR design basis.

Therefore, the staff will require that the multiple SGTR analysis be included as part of design safety analysis in Chapter 15 of the FDA/DC application. The objective of the analysis is to evaluate the concern that the response to multiple SGTR events in a passive PWR may result in unborated secondary water and present additional challenges to the passive safety systems. Therefore, the multiple SGTR events can be evaluated with a realistic analysis with careful consideration of operator actions and possible operator errors.

4.2 Secondary Water Chemistry

Section 4.2.7 of Chapter 3 requires that the plant design provide the system features that will facilitate compliance with the EPRI secondary water chemistry guidelines (EPRI Report NP-6239, "PWR Secondary Water Chemistry Guidelines," Revision 2, and its subsequent revisions). The steam generator design and materials will be selected to be consistent with those water chemistry requirements. Makeup water quality requirements will be those specified in Section 5 of Chapter 1 of the passive Requirements Document.

The staff concludes that these requirements are acceptable because the steam generator design and materials will be compatible with the secondary water chemistry.

4.3 System Features - Arrangement

Steam generators in the advanced PWR will be mounted at an elevation above the reactor vessel elevation to permit draining of the steam generator primary side (for tube inspection and plugging) without lowering the water level in the reactor vessel below the reactor core or adversely affecting shutdown cooling flow. Reactor coolant nozzle dams will be designed to facilitate handling and to minimize installation time. These nozzle dams will be designed so that they can be installed and removed using robotics, thereby minimizing personnel dose. Since these features are intended to minimize doses to personnel in accordance with the guidelines of RG 8.8, the staff finds them to be acceptable.

4.4 Component Features

Section 4.4.1 of Chapter 3 of the passive Requirements Document specifies the following requirements:

- General requirements applicable to steam generator materials are found in Section 5 of Chapter 1 of the passive Requirements Document. In addition, materials for tubes, tube supports, feed ring, tube sheet, and other internal parts will provide adequate resistance against damaging effects such as wastage and thinning of tubes, denting, erosion-corrosion resulting from thermal-hydraulic conditions and corrosive chemicals, intergranular attack, stress corrosion cracking, tube pitting and cracking, tube fretting and other known degrading factors. Wetting/drying action will be considered in selecting shell material.

- The steam generator is a critical component as defined in Section 5 of Chapter 1, and all the materials used in the steam generator and their heat treatment are to be identified. Vendors will document the heat and lot designation of each tube in the steam generator and provide archive samples of each heat and lot to the utility.

- The reference steam generator tube material will be annealed and thermally treated Ni-Cr-Fe Alloy 690. Before the final selection of the tube material, the plant designer will review operating experience and laboratory data for steam generator tube materials to confirm the tube material and processing selected. Alloy 690 steam generator tubing will be fabricated in accordance with the guidelines of EPRI Report NP-6743L, "Guidelines of Alloy 690 Thermally Treated." The reference processing for the tubes will be as follows:

  - A high-temperature solution anneal with a temperature and hold time that ensures that all carbon is placed in solid solution during final anneal will be used.

  - After rotary straightening and belt polishing, the tubing will be thermally treated at between 1320 and 1360 °F for a minimum of 5 hours.

  - U-bend tubes with bend radii small enough to be accommodated by the largest available vacuum furnace will be stress relieved for 2 hours at 1300 °F after the bend is formed.

- The concentration of cobalt in the steam generator tubes will be less than 0.015 weight percent. This will reduce the amount of activated cobalt in the reactor coolant system and reduce the overall dose rates.

- The tube supports will be constructed of a material demonstrated by steam generator service or by steam generator model testing under realistic flow conditions and off-design secondary water chemistry as providing adequate support (i.e., no fretting) and minimum potential for denting (i.e., deformation of tubes by corrosion of support materials). Stainless steel (300 Series and 400 Series) material is considered acceptable for tube support material. Carbon steel will not be used for tube support material.

- Tube bundle support materials and geometry will minimize fretting wear or galvanic corrosion of the steam generator tubes. The surface finish of support plate holes will not produce excessive tube wear over the design life of the steam generator. Circular drilled holes will not be used for

tube support material. The tube support geometry will be demonstrated by steam generator service or by steam generator model testing under realistic flow conditions and off-design water chemistry as providing adequate support and minimal potential for denting.

- Steam generator tube-to-tube sheet joints will incorporate tube-to-tube sheet seal welds at the primary side of the tube sheet and full expansion of the tubes into the tube sheet for the full thickness of the tube sheet so that crevices are eliminated. The tube expansion equipment and procedure will be shown to minimize residual stress in the transition from the expansion zone to the unexpanded tube.

- Measures will be taken to ensure that tubes are not expanded above the secondary side of the tube sheet.

- Clear access, including platforms, will be provided to facilitate inspection and maintenance of steam generator tubes. Primary channel head manholes will be a minimum of 21 inches in diameter. Remotely operated stud tensioning and detensioning devices will be provided to minimize the time spent by maintenance personnel in high dose rate areas when removing or installing these steam generator manhole covers.

The staff concludes that these requirements are acceptable. The materials selection and fabrication requirements in this section and in Section 5 of Chapter 1 will ensure that the steam generator will perform in service as designed.

4.5 Steam Generator Water Level Control

Section 4.5.1 of Chapter 3 specifies the requirements for steam generator water level control. Appendix B to Chapter 1 indicates that the ALWR will be in compliance with the guidance of GL 89-19, "Request for Action Related to Resolution of Unresolved Safety Item A-47, 'Safety Implication of Control Systems in LWR Nuclear Power Plants,' pursuant to 10 CFR 50.54(f)," which requires each PWR plant to provide an automatic steam generator overfill protection system that is sufficiently separated from the main feedwater control system. Section 7.13 of Chapter 10 of the passive Requirements Document requires the man-machine interface systems for the steam generator system to provide the monitoring and control necessary to maintain an adequate inventory so that overfilling a steam generator can be prevented. In its letter dated May 13, 1991, EPRI stated that the details of the method for providing the required protection are the responsibility of the man-machine interface system (M-MIS) designer and must be provided as part of the overall M-MIS design. Therefore, the staff will evaluate the specific design during its review of an individual application for FDA/DC.

4.6 Steam Generator Ventilation

Section 4.6.1 of Chapter 3 requires that the reactor coolant system arrangement allow sufficient space in the vicinity of the steam generator primary heads for the installation and use of temporary ventilation exhaust systems during personnel access into the steam generator primary side. The containment purge exhaust duct will have a permanent connection for flexible

exhaust ducting from the steam generator primary side. This feature will minimize the potential for the inhalation of airborne radioactivity in the steam generator primary side, and the staff finds it acceptable.

5 BWR REACTOR COOLANT SYSTEM

The BWR RCS will transport the main steam from the reactor vessel through the outmost main steam isolation valve and feedwater from the feedwater system to the reactor vessel. Section 5.1.2 of Chapter 3 of the passive Requirements Document specifies the functional requirements for the BWR RCS.

5.1 BWR Interfacing System LOCA (ISLOCA)

Section 5.1.3 of Chapter 3 identifies the systems that will interface with the BWR RCS: RWCU systems, control rod drive (CRD) system, depressurization valve, feedwater system, passive decay heat removal system (PDHRS), and passive safety injection system (PSIS). In Table II of its May 13, 1991, response to a staff letter dated February 28, 1991, EPRI provided its ISLOCA evaluation of various interfacing systems. EPRI concluded that the design of these systems and subsystems interfacing with the BWR RCS will be in full compliance with the requirements for ISLOCA protection as specified in Section 2.2.14 of Chapter 3. However, a certain portion of these systems does not meet the RCS full-pressure ultimate rupture strength (URS) criterion. These lower pressure portions are the following:

- The RWCU system outboard lines to the condenser and radwaste vents and drain are designed for low-pressure subsystems.
- The portion of the CRD system from the pump suction to the condensate storage tank is designed for atmospheric pressure plus hydraulic head.
- The feedwater system piping from the pump suction is designed for process conditions.
- The PDHRS piping beyond the vent and drain isolation valves is designed for process conditions.

EPRI contends that these systems are beyond the "extent practicable" definition of Section 2.2.14.1 of Section 3, but does not give bases to justify the impracticality of designing them to meet the RCS full-pressure URS criterion.

As stated in Section 2.1.7 of Chapter 3 of this report, the staff's position is that, to the extent practicable, piping runs and associated elements should be designed to meet the RCS full-pressure URS criterion, and that the degree of isolation is not sufficient justification for using low-pressure components. For all interfacing systems and components that do not meet the RCS full-pressure URS criterion, the staff requires that the plant designer justify the impracticality of designing a low-pressure system to the URS criterion to reduce the pressure challenge. While reviewing an individual application for FDA/DC, the staff will further evaluate the design of the interfacing systems, including justifications for impracticality if a system component is not designed to meet the RCS full-pressure URS criterion.

5.2 Water Chemistry

Section 5.2.2 of Chapter 3 specifies that the RCS design will be compatible with operating and layup conditions in accordance with the water quality

specifications in Section 5 of Chapter 1 of the passive Requirements Document. These specifications provide the environmental definitions to be used for selecting system design and material and will form the basis for fouling factors for heat transfer surfaces, for the limiting of carryover of volatile components in the steam, and for the minimization of transport and deposition of radioactive materials.

The staff concludes that these requirements are acceptable. Water quality specifications will ensure that corrosion is kept under control, especially intergranular stress corrosion cracking of materials. The water chemistry limitations are included in Section 5 of Chapter 1, which the staff found acceptable.

5.3 System Features

Section 5.3.2 of Chapter 3 specifies the following requirements:

- Materials selected will be compatible with the appropriate water chemistry conditions for standard and non-standard conditions as specified in Chapter 1 of the passive Requirements Document.

- Specific materials to be used in the RCS are as follows:

  - main steamlines - carbon steel

  - feedwater lines - carbon steel

  - main steam isolation valves (MSIVs) and safety/relief valves (SRVs) - carbon steel body

  - differential pressure valves - carbon steel, Type 304L or 316L body; if no welding is done on the primary pressure boundary, the material may be Type 304 or 316 stainless steel.

- SRV discharge piping material will be selected so that the suppression pool water quality limits in Chapter 1 will be met.

The staff concludes that these requirements are acceptable because these materials have performed satisfactorily in existing nuclear plants.

5.4 Main Steamlines

Section 5.3.3.3 of Chapter 3 states that a separate MSIV leak detection and control subsystem will not be provided for the ALWRs. EPRI identifies this issue as a plant optimization subject and discusses it in Section 2.3.1 of Appendix B to Chapter 1 of the passive Requirements Document. The staff documents its evaluation in Section 2.3.1 of Appendix B to Chapter 1 of this report.

Steam Relief

Section 5.3.4.6 of Chapter 3 requires instrumentation to be provided for safety/relief valve (SRV) leakage detection. In the corresponding rationale portion, EPRI originally stated that "leakage monitoring is specified so that

SRV leakage does not contribute to the containment undetected leakage limits." In its letter dated May 13, 1991, regarding whether leakage detection should also allow for leakage quantification, EPRI stated that, because the SRV leakage will be past the valve seats and into the suppression pool through the SRV discharge piping, it cannot contribute to unidentified leakage in the containment. Therefore, EPRI revised this section of the passive Requirements Document in Revision 2 by deleting the reference to the containment undetected leakage limits. EPRI revised the corresponding rationale portion to state that "leakage monitoring is specified so that the need for maintenance is known before leakage becomes excessive." The staff concludes that this revision does not conflict with any regulatory requirements and is, therefore, acceptable.

5.5 Feedwater System

Section 5.3.5.2 of Chapter 3 states that each feedwater line penetrating the primary containment will be equipped with a simple check valve inside the containment and a positive acting check valve, with spring-assisted seating, outside the containment. A third remotely operable gate valve with high leaktight capability is required for each feedwater line upstream of the isolation valve outside the containment. EPRI initially proposed that the third valve need not meet the requirements of Appendix J to 10 CFR Part 50. The check valves are specified as the isolation valves. The third valve is provided for positive shutoff when the feedwater system is not in service to permit maintenance on the upstream feedwater components and to facilitate leak testing of check valves in accordance with Appendix J to 10 CFR Part 50.

In the DSER for Chapter 3, the staff concluded that it was not acceptable to exclude the third valve from the leak testing requirements in Appendix J to 10 CFR Part 50. It stated that since the above valve arrangement was similar to the typical feedwater line valve arrangement described in American Nuclear Society (ANS) 56.2-1976 as satisfying General Design Criterion 55 of Appendix A to 10 CFR Part 50, the valve arrangement was acceptable. However, as described in Note 55-1 of ANS 56.2-1976, all three of the valves in the above valve arrangement were considered containment isolation valves. Therefore, Appendix J leak testing was required for all three valves. EPRI's proposal, which did not require the third valve to be Appendix J leak tested, was not acceptable. The staff identified this as an open issue in the DSER. Thus, in Revision 4, EPRI revised Section 5.3.5.2 of Chapter 3 to include a third valve with high leaktight capability. The staff reviewed these changes and concludes that they are acceptable. Therefore, this open issue is closed.

Section 5.3.5.5 of Chapter 3 requires flow and temperature measurement devices to be provided in each feedwater line for the feedwater/level control system. GL 89-19 requires that each BWR plant design be provided with automatic reactor vessel overfill protection to mitigate main feedwater overfeed events and that the overfill protection system be sufficiently separated from the main feedwater (MFW) control system to ensure that the MFW pumps trip on a reactor high-water-level signal. Though Appendix B to Chapter 1 of the passive Requirements Document indicates EPRI commitment that ALWRs will comply with GL 89-19, the details of compliance will be developed by the plant designer and are beyond the scope of the passive Requirements Document. The staff will evaluate the specific design during its review of an individual application for FDA/DC.

EPRI stated that it would add a new section (Section 5.3.5.8) to address the mitigation of anticipated transients without scram (ATWS). Because there are no recirculation pumps in the passive BWR design for power control, there is one less negative reactivity insertion method than that required by 10 CFR 50.62. The staff expressed its concern about the effect of continued natural circulation on the consequences of ATWS events. In its letter dated May 22, 1991, EPRI indicated that automatic feedwater runback will be the negative insertion method in lieu of the recirculation pump trip, and that the effect of natural circulation will be included by the plant designer in the analysis of the ATWS events. In the DSER for Chapter 3, the staff stated that EPRI had committed to add these requirements to the passive Requirements Document as Section 5.3.5.8 of Chapter 3. This was identified as a confirmatory issue in the DSER. Revision 3 included a new requirement in Section 4.2.3.8 of Chapter 2, requiring that the automatic runback be provided in conjunction with the reactor protection system to reduce the transient effects on the core and RCS boundary. Therefore, the DSER confirmatory issue is closed. The staff agrees that automatic feedwater runback is an acceptable means of negative reactivity insertion. However, the effect of natural circulation and acceptability of the consequences of the ATWS events will be evaluated as a part of the design-specific ATWS analysis. Therefore, the requirement for the plant designer to include the effect of natural circulation in the analysis of ATWS events is acceptable. The staff will evaluate the analysis during its review of an individual application for FDA/DC.

The staff concludes that the above requirements, with the exception of the requirements for ATWS noted above, do not conflict with NRC requirements and regulatory guidelines.

6 CHEMICAL AND VOLUME CONTROL SYSTEM (PWR AUXILIARY)

The chemical and volume control system (CVCS) will maintain the RCS inventory and control its chemistry. Section 6.1.2 of Chapter 3 of the passive Requirements Document specifies the following CVCS functional requirements: (1) maintain required water inventory and control boron concentration in the RCS during normal operation and normal shutdown, (2) provide means for chemical and hydrogen additions, (3) demineralize and filtrate reactor coolant, (4) remove noble gases from the reactor coolant letdown stream, (5) provide borated makeup water for the spent fuel pool and the refueling water storage tank, (6) provide a reactor coolant makeup water supply to various auxiliary equipment, and (7) provide makeup water for losses from small leaks in the RCS piping. The CVCS is also required to supply pressurizer auxiliary spray water for depressurization.

Section 6 of Chapter 3 specifies that the selection of materials for the charging pumps, heat exchangers, boric acid tanks, boric acid batching tank, flow element control, and valves must meet the general materials requirements in Section 5 of Chapter 1 of the passive Requirements Document. The materials selection criteria in Section 5 of Chapter 1 were reviewed by the staff and found acceptable.

Section 6.2.1 of Chapter 3 specifically states that the CVCS will not be required to perform any safety-related functions, such as accident mitigation or safe shutdown; therefore, requirements applied to safety-grade systems will not apply to the design of the CVCS. This non-safety-grade requirement results in a simplified CVCS design in regard to equipment qualification, physical separation, redundancy for single-failure consideration, power supplies, instrumentation requirements, plant technical specifications, and surveillance requirements. The passive Requirements Document also requires that the simplification of the CVCS not compromise the defense-in-depth functions as the normal means of maintaining RCS inventory and primary water chemistry. In the corresponding rationale portion, EPRI indicates that system reliability and availability can be enhanced through design improvements, judicious location of components for better inservice environments and improved capabilities, and selected application of redundancy and diversity requirements.

Section 6.2.2.2 of Section 3 originally required that a boron dilution be limited so that the operator has a minimum of 30 minutes after the beginning of an unintended boron dilution to identify and terminate the incident before criticality is reached during any refueling operation. In a letter dated February 28, 1991, the staff stated that this requirement did not account for the time necessary for instrumentation and alarms to respond. In Revision 2, EPRI revised the requirement as follows: "The operator has a minimum of 30 minutes after an indication of an unintended boron dilution to identify and terminate the incident prior to reaching criticality during any refueling operation." In Revision 4, EPRI added another statement to the rationale section, acknowledging that the operator must have the capability of monitoring core reactivity conditions at shutdown in order to identify and terminate boron dilution events.

Since the CVCS of the passive plant is not required to perform any safety-related function, many requirements of SRP Section 9.3.4, "Chemical and Volume Control System (PWR) Including Boron Recovery System," regarding the CVCS safety functions do not apply. For example, boration of the RCS and RCS inventory control for accident mitigation or for achieving safe shutdown is accomplished by the safety-grade passive safety injection system. However, in performing the licensing design-basis (LDB) analysis, the plant designer must demonstrate that a decrease in boron concentration in the reactor coolant caused by mal-functions of the CVCS will not result in the specified design limits being exceeded. This LDB analysis requirement is properly included as Event 4.5 in Table 1.2-1, Chapter 1, of the passive Requirements Document and is acceptable.

As the non-safety-grade CVCS is relied on to perform defense-in-depth functions, its reliability/availability missions will be determined as part of the regulatory treatment of non-safety systems discussed in Section 2.2 of this chapter. During its review of individual applications for FDA/DC, the staff will evaluate the system design capability to maintain RCS water chemistry and makeup during normal operation and other aspects of reliability assurance and operational control to verify that they are consistent with the reliability/availability missions from EPRI's process for determining the regulatory treatment of non-safety systems.

7 POSTACCIDENT SAMPLING

Section 7.2.8 of Chapter 3 of the passive Requirements Document contains requirements for the design of the postaccident sampling system (PASS). Additional information concerning the PASS is provided in Section 2.3.2 of Appendix B to Chapter 1 of the passive Requirements Document. The original version of the passive Requirements Document deviated from the requirements of Item II.B.3 of NUREG-0737, "Clarification of TMI Action Plan Requirements," in that EPRI originally proposed five exceptions to the requirements for the PASS. In the DSER for Chapter 3, the staff identified the simplification of the PASS as an open issue and included it as a policy issue in the February 27, 1992, draft Commission paper.

As described in Section 2.3.2 of Appendix B to Chapter 1 of this report, the staff concludes that EPRI's requirements for the PASS, except for the deletion of sampling for dissolved gases in PWRs, are acceptable. The staff will review individual applications for FDA/DC to ensure that passive plant vendors meet the requirements of Item II.B.3 of NUREG-0737, with the deviations approved by the Commission in its SRM dated July 21, 1993. Therefore, this open issue is closed.

System Features - Arrangement

To obtain representative liquid and gaseous samples (with minimum radioactive decay and plateout), line lengths to sample stations will be minimized. Sample station components carrying potentially radioactive fluids will be located behind a shield wall, and area dose rates in sampling stations will be maintained as low as is reasonably achievable (ALARA) to minimize personnel doses. These features are in accordance with the guidelines of RG 8.8, and are acceptable.

Materials

Section 7.4.1 of Chapter 3 of the passive Requirements Document requires that the material requirements be in accordance with Section 5 of Chapter 1. The system must not contribute to sample contamination by such mechanisms as cobalt or corrosion-product release. The staff concludes that these requirements are acceptable. The staff reviewed Section 5 of Chapter 1 and found it acceptable.

8 BWR REACTOR WATER CLEANUP SYSTEM

Section 8 of Chapter 3 of the passive Requirements Document requires the reactor water cleanup (RWCU) system to provide continuous reactor coolant cleanup under normal and transient conditions. Section 8 also includes functional requirements that the RWCU system provide (1) a method of removing and processing primary coolant from the reactor during normal operation and (2) a flow path from the reactor pressure vessel (RPV) bottom head to give assurance that neither crud nor cold water will accumulate in the lower vessel head area. These specifications for the RWCU system meet SRP Section 5.4.8, "Reactor Water Cleanup System (BWR)."

Section 8 also requires the RWCU system to have heat removal capacity that will provide a backup to the passive decay heat removal system during periods of reactor isolation and during cooldown and shutdown. This is a new operating requirement for the RWCU system. The performance requirements for the RWCU shutdown cooling function are contained in Section 9.2.1 of Chapter 3, and are common for both PWRs and BWRs. Since the non-safety-grade RWCU system is required to perform defense-in-depth functions, its reliability and availability requirements will be addressed consistent with the staff review discussed in Section 2.2 of this chapter. In addition, the staff will review the shutdown risk issue as discussed in Section 9 of this chapter.

Section 8.2.2 of Chapter 3 of the passive Requirements Document specifies that the system will remove insoluble colloidal and soluble ionized impurities to

- maintain reactor water quality by not exceeding the RWCU effluent specification in Chapter 1
- reduce iodine concentrations in reactor coolant before and during shutdown but before the RPV head is removed
- deliver water of low radioactivity to the main condenser or to radwaste

The staff concludes that these requirements are acceptable. Removal of impurities will maintain the reactor water quality as specified in Section 5 of Chapter 1 of the passive Requirements Document.

The RWCU system (BWR) will be shielded so it can continue operation during shutdown and not be a significant radiation source to other shutdown maintenance activities. Valves and instruments will be separated from high-radiation sources, and pumps, filters, and demineralizers will be separately shielded. Areas requiring personnel access such as operating panel filter/demineralizer resin makeup/replacement areas and sample stations will be accessible and located in low-radiation areas. Equipment requiring maintenance will be accessible and provided with adequate spacing. All of the above features are in compliance with the guidelines of RG 8.8 and are intended to maintain personnel exposures ALARA. The staff, therefore, finds them to be acceptable.

Section 8.4.1 of Chapter 3 of the passive Requirements Document states that the general material requirements applicable to the RWCU system are as described in Section 5 of Chapter 1. All system components will be selected considering the following:

- freedom from general and localized corrosion
- limiting of corrosion-product formation and release and effect on water quality
- minimizing of radiation buildup

The staff concludes that these requirements are acceptable. The staff reviewed Section 5 of Chapter 1 and found it acceptable.

9 PWR REACTOR SHUTDOWN COOLING SYSTEM

Section 9 of Chapter 3 of the passive Requirements Document states that the reactor shutdown cooling (RSDC) system will be designed to perform the following functions: (1) have sufficient capacity to remove core decay heat and RCS sensible heat during normal plant cooldowns in a time consistent with the refueling schedule described in Section 9.2.1.2.1, (2) remove core decay heat during refueling and maintenance outages, (3) provide a backup to the passive containment cooling system for removing core decay heat in the long-term cooling following a loss-of-coolant accident (LOCA), (4) transfer reactor coolant to the chemical and volume control system (CVCS) for purification when the RCS is at reduced pressure and the reactor coolant pumps are stopped (PWR), (5) be capable of removing heat from and preventing steaming of the in-containment refueling water storage tank during passive residual heat removal (RHR) system operation (PWR), and (6) provide the relief valves for low-temperature overpressure protection for the RCS.

Section 9.3.8.3 of Chapter 3 requires the capability for testing the RSDC isolation valve leakage rate, and Section 9.3.13.1 requires monitoring capability and alarms for certain parameters including the RSDC pump suction and discharge pressures. The staff raised a concern that the passive Requirements document included no requirements for valve position indication and high-pressure alarms in the control room to warn the operators when rising RCS pressure was approaching the design limits of the RSDC system. EPRI resolved this concern in Revision 2 of the passive Requirements Document by adding Section 2.2.14 in Chapter 3, which requires valve position indication and high-pressure alarms in the control room.

Section 9.3.9.2 of Chapter 3 requires the RSDC pump seals to accommodate operating pressures of up to 900 psig and to be capable of withstanding full RCS pressure without seal leakage exceeding normal CVCS charging capability. EPRI indicates that the 900-psig operating pressure satisfies the staff position on interfacing system LOCA protection that requires pump seals of the interfacing systems to be designed to the ultimate rupture strength that is at least equal to the RCS pressure. Since no design details are available, the staff will evaluate this issue during its review of an individual application for FDA/DC. The staff's position regarding intersystem LOCA is discussed further under Item I.F of Annex A of Appendix B to Chapter 1 of this report.

Shutdown and Low-Power Operation Risk

Various incidents occurring at nuclear power plants during low-power and shutdown operation modes over the past several years have been a concern to the NRC staff. The shutdown events have caused plants to lose the capability to maintain core cooling, provide makeup coolant to the reactor, and maintain electrical power to essential equipment. The event at the Diablo Canyon Nuclear Power Plant on April 10, 1987, emphasized the sensitivity of operating a PWR with a reduced inventory in the RCS. On October 17, 1988, the staff issued Generic Letter (GL) 88-17, "Loss of Decay Heat Removal," in which it requested licensees to address numerous generic deficiencies to improve safety during operation at reduced inventory. Recently, in the report on the loss of power at the Vogtle plant, NUREG-1410, "Loss of Vital AC Power and the Residual Heat Removal System During MidLoop Operations at Vogtle Unit 1 on

March 20, 1990," issued in May 1990, the NRC incident investigation team discussed the need to manage risk for shutdown operations. In February 1992, the staff issued NUREG-1449, "Shutdown and Low Power Operation at Commercial Nuclear Power Plants in United States," to address risk associated with shutdown conditions.

The staff reviewed ALWR requirements for midloop operations on the basis of insights from NUREG-1410; NUREG-1449; NUREG/CR-5820, "Consequences of the Loss of the Residual Heat Removal Systems in Pressurized Water Reactors," issued in April 1992; and NUREG/CR-5819, "Probability and Consequences of Rapid Boron Dilution in a PWR," issued in March 1992.

Section 9.2 of Chapter 3 identifies performance requirements for the RSDC system during normal shutdown cooling and midloop operation. Section 9.2.1.2 requires that the RSDC system be designed to provide sufficient decay heat removal capability to cool RCS water from the RSDC system entry temperature to refueling temperature, and to maintain this temperature until plant startup to completion of the refueling outage. The RSDC system will also be designed to bring the reactor to a cold shutdown condition within 36 hours and will prevent the RCS temperature from returning to boiling once the reactor vessel head has been removed.

Section 9.2.2.1.1 requires that all potential RSDC system conditions be analyzed and properly account for sources of errors during midloop operations to ensure that the plant design will be able to maintain core cooling. The ALWR analyses for PWR midloop operations require the determination of the minimum water level for maintenance and continued RSDC system operation and the demonstration of the adequate RCS level with minimal or no restrictions on system flow rate. The incipient vortex formation level will be defined, and it will be verified that this level is below the minimum defined RCS level to prevent vortex formation and air entrapment in the suction of the RSDC system pumps. Adequate net positive suction head (NPSH) for the RSDC system suction pumps will be ensured and verified for all shutdown conditions. The minimum leakage detection time based on leakage, location, and detection method will be determined, and it will be verified that the inventory lost will be detected in time to permit corrective actions to prevent a loss of the RSDC system.

Although the most significant events have occurred at PWRs, the potential vulnerability of BWR plants can be vulnerable to shutdown and low-power events. Section 9.2.3 of Chapter 3 specifies normal shutdown cooling performance requirements for BWRs. Section 9.2.3.1.1 requires that potential RWCU conditions be analyzed and that significant sources of errors be accounted for. The analyses required for reduced inventory conditions in BWRs include the determination of the minimum water level for maintenance in all shutdown conditions and continued RWCU operation. This minimum RCS water level will be verified, and it will be demonstrated that adequate flow rate exists for continued RWCU operation with minimal or no restrictions on systems. Analyses for vortex formation, adequate NPSH for pump suction, and leakage detections are also required, similar to the requirements for PWR designs.

Section 1.1.4 of Appendix A to Chapter 1 requires that a limited-scope of probabilistic risk assessment (PRA) study be conducted for shutdown and low-power operation to determine which sequences can lead to core damage, to quantify and identify the potential vulnerabilities, and to remedy the most important ones. The staff agrees that the PRA study will provide insights into the important contributors associated with other operating conditions.

The passive Requirements Document includes new requirements to address concerns regarding the capability to cool the core on loss of RSDC with upper internals in place. Section 6.3.2.3.5 of Chapter 4 was added to require that the PWR upper internals be sufficiently designed for adequate natural circulation flow between the reactor vessel and the refueling pool, and to prevent void formation under the upper internals with the RSDC system not operating. Sections 8.2.1.1 and 8.2.2.3 of Chapter 1 were added to provide additional assurance for shutdown operation by requiring the plant designer to perform best-estimate analyses to identify significant shutdown risk concerns and provide a basis for preparing normal and emergency procedures, technical specifications (TS), and operator training material. In the analysis, the designee will identify potential passive safety issues associated with events that might occur during shutdown, plant configurations that might be present and affect the severity of a shutdown accident, and passive plant features that will be made available to prevent or mitigate the effects of the potential events.

Also, the appropriate emergency procedures requirements will be determined for RCS inventory control, shutdown reactivity control, containment isolation (including isolation time requirements with fuel in the vessel), and requirements for decay heat removal and electrical power. Further, the emergency procedures developed for use during shutdown and low-power operation will have procedure entry conditions unique to shutdown conditions and low-power operation.

EPRI's responses to shutdown risk concerns reflect broad requirements for shutdown and low-power-operation conditions. The requirements address concerns from a variety of sources including foreign risk assessment, plant operating experiences, and NUREG-series reports.

There is one remaining issue about which the staff is concerned regarding the EPRI shutdown requirements to provide an independent RCS level measurement method. In response to the staff questions regarding various shutdown risk concerns, EPRI stated in its letter dated December 2, 1991, that the passive Requirements Document requires that a redundant and independent RVLIS be provided to operators to monitor reactor water level from the control room throughout the shutdown evolution. The staff reviewed Section 6.3.3.1 of Chapter 4 and found that the passive Requirements Document requires only redundant, but not independent, level measurements. GL 88-17 specifically requires two independent RCS level measurements during shutdown operations. The staff will evaluate individual application for FDA/DC to ensure that two independent RCS level measurements are provided during shutdown operations.

The staff concludes that EPRI's requirements for midloop and shutdown operation are acceptable because all potential RSDC system conditions will be analyzed and sources of errors during midloop operations will be accounted for, including the determination of the minimum water level for continued RSDC

system operation to preclude potential loss of shutdown cooling because of vortex formation and air entrapment. In the analysis, the designer will identify potential passive safety issues associated with events that might occur during shutdown, plant configurations that might be present and affect the severity of a shutdown accident, and passive features that will be made available to prevent and mitigate the effects of the potential events.

During the review of an individual application for FDA/DC the staff will verify that the plant designer has properly addressed the following items: appropriate implementation of the GL 88-17 requirements, adequate TS requirements for shutdown operations, adequate procedural guidance for coping with a loss of the decay heat removal system, capability to maintain core cooling in all shutdown conditions, loss of RCS inventory, outage planning and control program for shutdown operations, potential boron dilution associated with low-power operation and shutdown conditions, and design enhancements to prevent or mitigate the effects of a loss of RSDC system pump suction when the RCS level is lowered.

The staff has not yet determined the appropriate requirements pertaining to shutdown risk. The staff is assessing its resolution of issues identified in NUREG-1449. If these studies result in new regulatory and operational requirements, they will be implemented as part of the FDA/DC and COL reviews.

Chemistry Control

Section 9.3.7 of Chapter 3 specifies that means will be provided to permit the sampling of the boron concentration in the RSDC system before initiating system operation during plant cooldowns and following system maintenance, and to adjust the boron concentration if required. Sampling capability will also be provided for abnormal RCS conditions (i.e., with high fission product activity in the coolant).

The staff concludes that these requirements are acceptable because they will ensure that the operation of the RSDC system will not result in boron dilution in the RCS.

10 CONCLUSION

The staff concludes that the EPRI requirements established in Chapter 3 of the passive Requirements Document for the design of the reactor coolant system and reactor non-safety auxiliary systems do not conflict with current regulatory guidelines and are acceptable. However, by themselves they do not provide sufficient information for the NRC staff to determine that the plant-specific design, operation, and arrangement of the reactor coolant system and reactor non-safety auxiliary systems will be adequate. Applicants referencing the passive Requirements Document will be required to demonstrate compliance with the additional guidance provided in the SRP (NUREG-0800), or provide justification for alternative means of implementing the associated regulatory requirements.

Therefore, the staff concludes that Chapter 3 specifies requirements that, subject to the resolution of the identified vendor- and utility-specifications, if properly translated into a design in accordance with the NRC regulations in force at the time the design is submitted, should result in a nuclear power plant whose reactor coolant system and reactor non-safety auxiliary systems will perform as designed and have all the attributes required by the regulations to ensure that there is no undue risk to the health and safety of the public or to the environment.

APPENDIX A
DEFINITIONS AND ACRONYMS

Appendix A of Chapter 3 of the passive Requirements Document contains definitions of acronyms. The staff has provided a consolidated list of acronyms in Volume 1 of this report.

CHAPTER 4, "REACTOR SYSTEMS"

1 INTRODUCTION

This chapter of the SER documents the NRC staff's review of Chapter 4, "Reactor Systems," of the passive Requirements Document through Revision 5. Revision 5 of the passive Requirements Document was submitted to the NRC by letter dated October 1, 1993. Chapter 4 was prepared, under the direction of EPRI and the ALWR Utility Steering Committee, by General Electric Company; S. Levy Incorporated; MPR Associates, Inc.; Westinghouse Electric Corporation; and EPRI.

1.1 Review Criteria

Section 1 of Volume 1 of this report describes the review criteria used by the staff during its review of Chapter 4 of the passive Requirements Document.

1.2 Scope and Structure of Chapter 4

Chapter 4 of the passive Requirements Document defines the ALWR Utility Steering Committee's requirements for the design of the reactor systems at ALWRs in which passive safety systems will be used.

The key topics addressed in the Chapter 4 review include BWR reactor pressure vessel (RPV) and internals, BWR core and fuel, BWR control rod drive (CRD) system, PWR RPV and internals, PWR core and fuel, and PWR CRD system.

1.3 Policy Issues

During its review of Chapter 4 of the passive Requirements Document, the staff did not identify any issues that involve policy questions for the technical areas discussed in this chapter, other than those already identified in the Commission papers listed in Appendix B to Chapter 1 of this report.

1.4 Outstanding Issues

The DSER for Chapter 4 of the passive Requirements Document contained the following open issues:

Open Issues

P.4.0-1 rod insertion capability after an earthquake (2.2.6)
P.4.0-2 inservice inspection of RPV internals (2.3.2)
P.4.0-3 RPV thermocouples (3.3)

Confirmatory Issues

None

The final disposition of the open issues is discussed in detail in the appropriate sections of this chapter, as indicated by the parenthetical notation following each issue. All outstanding issues identified in the DSER for Chapter 4 have been resolved.

1.5 Vendor- or Utility-Specific Items

The DSER for Chapter 4 contained the following vendor- or utility-specific items. The final disposition of each item is discussed in the appropriate section of this chapter, as indicated by the parenthetical notation following each item.

P.4.V-1 boration of water for fuel handling and storage (2.2.8)
P.4.V-2 fuel assembly reconstitution (2.3.3)
P.4.V-3 condensation carryunder limitation (3.2)
P.4.V-4 experimental data for divided-chimney design (3.2 and 4.2.1)
P.4.V-5 BWR stability (2.2.4 and 4.2.1)
P.4.V-6 decay ratio limits and analysis methods and procedures (4.2.1)
P.4.V-7 loss-of-coolant-accident analysis methodology (4.2.1 and 7.2.1)
P.4.V-8 load-following and maneuvering capability (4.2.1 and 7.2.1)
P.4.V-9 methodology to achieve BWR stability (4.2.1)
P.4.V-10 fuel burnup requirements (4.2.2 and 7.2.2)
P.4.V-11 two-cycle fuel channel lifetime (4.2.4)
P.4.V-12 control rod assembly lifetime (4.2.6 and 7.2.3)
P.4.V-13 control rod scram time (5.2)
P.4.V-14 control rod assembly malfunctions in BWR accident analyses (5.2)
P.4.V-15 protection of scram pilot solenoid valves (5.3)
P.4.V-16 thermal shield removal (6.3)
P.4.V-17 RPV level instrumentation (6.3)
P.4.V-18 negative moderator temperature coefficient limit (7.3)
P.4.V-19 fuel rod bow penalties (7.3)
P.4.V-20 CRD mechanism lifetime (8.2)

The following vendor- or utility-specific items remain and will be addressed during the staff's review of an individual application for FDA/DC. The staff's position on each item is provided in the appropriate section of this chapter, as indicated by the parenthetical notation following each item.

P.4.V-1 boration of water for fuel handling and storage (2.2.8)
P.4.V-2 fuel assembly reconstitution (2.3.3)
P.4.V-3 condensation carryunder limitation (3.2)
P.4.V-4 experimental data for divided-chimney design (3.2)
P.4.V-5 BWR stability (4.2.1 and 4.2.2)
P.4.V-6 decay ratio limits and analysis methods and procedures (4.2.1)
P.4.V-7 loss-of-coolant-accident analysis methodology (4.2.1)
P.4.V-8 load-following and maneuvering capability (4.2.1 and 7.2.1)
P.4.V-9 two-cycle fuel channel lifetime (4.2.4)
P.4.V-10 control rod scram time (5.2)
P.4.V-11 control rod assembly malfunctions in BWR accident analyses (5.2)
P.4.V-12 protection of scram pilot solenoid valves (5.3)
P.4.V-13 thermal shield removal (6.3)
P.4.V-14 RPV level instrumentation (6.3)
P.4.V-15 fuel rod bow penalties (7.3)
P.4.V-16 CRD mechanism lifetime (8.2)

2 POLICY STATEMENTS AND KEY REQUIREMENTS

Sections 1 and 2 of Chapter 4 of the passive Requirements Document define the utility requirements for the passive ALWR that are common to BWR and PWR reactor systems. Section 1 specifies that the reactor systems are to (1) generate the thermal power necessary to meet required plant electrical power output while not exceeding specific nuclear, thermal-hydraulic, and mechanical design limitations; (2) serve as a pressure boundary and a barrier to prevent the release of radioactivity from the reactor core or reactor coolant; (3) provide a flow path for natural or forced circulation of coolant to remove heat generated by the reactor core under all operating conditions and to remove decay heat after shutdown; and (4) provide for control of core reactivity. These functional requirements are consistent with those for the current generation of LWRs and the staff finds no inconsistency with NRC regulations.

2.1 Reactor Pressure Vessel Reliability

Section 1.5.2 of Chapter 4 of the passive Requirements Document states that the reactor vessel is considered to be a critically important component of an ALWR plant. Its reliability and longevity are essential to safe and economic operation of an ALWR plant over its 60-year life. Accordingly, use of the best-available, proven technology is being required for construction of the ALWR RPV.

Ring-forged construction has been identified as the best-available, proven technology for reactor vessels. It has been applied to both PWR and BWR reactor vessels and provides the following benefits:

- The total number of welds in the vessel is substantially reduced.
- All longitudinal welds are eliminated from the vessel.
- All welds are eliminated from the highly irradiated beltline section of the vessel.

The reduced number of welds will reduce the amount of inservice weld inspection required. Longitudinal welds are considered more critical than circumferential welds because of higher stresses normal to the weld line. Eliminating welds from the beltline region enables all highly irradiated regions of the RPV to be forged material of optimum and well-defined chemistry and structure. Accordingly, the magnitude and variability of irradiation-induced shifts in nil-ductility temperature will be reduced, thereby simplifying low-temperature operations and providing added confidence that the reactor vessel can achieve a 60-year design life.

The staff concludes that these requirements are acceptable.

2.2 Performance Requirements

Section 2 of Chapter 4 of the passive Requirements Document specifies top-level generic performance and equipment design requirements common to both PWRs and BWRs.

2.2.1 Barriers Against Fission Product Release

Section 2.2.1 of Chapter 4 requires that the reactor systems be designed with two separate barriers against the release of fuel fission products: the fuel cladding and the reactor coolant system pressure boundary. This requirement is intended to establish the basic defense-in-depth principle to prevent the release of radioactive fission products to the environment and is acceptable.

2.2.2 Pressure Boundary Integrity

Section 2.2.2 of Chapter 4 requires that the RPV be designed and fabricated using proven technology. It also requires that RPV materials, design, and fabrication methods be selected so that the material will not behave in a brittle manner, be subject to rapidly propagating failure, or otherwise fail considering the environmental conditions that will be present during all normal operations and the events in the event frequency categories defined in Chapter 1 of the passive Requirements Document. The staff concludes that these requirements are intended to ensure RPV integrity and are, therefore, acceptable.

2.2.3 Negative Power Coefficient

General Design Criterion (GDC) 11 of 10 CFR Part 50, Appendix A, requires the reactor core and associated coolant system to be designed so that, in the power operating range, the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity. Section 2.2.3 of Chapter 4 of the passive Requirements Document requires the reactor to be designed to be self-limiting against uncontrolled power excursions under all operating conditions by requiring the power reactivity coefficient to be negative under all conditions. Because a negative power reactivity coefficient will result in a negative reactivity insertion as power increases, requiring a negative power coefficient meets the requirements of GDC 11. Therefore, the staff concludes that this requirement is acceptable.

2.2.4 Freedom From Power Oscillations

GDC 12 requires the reactor core and associated coolant, control, and protection systems to be designed to ensure that power oscillations that can result in conditions exceeding specified acceptable fuel design limits (SAFDLs) are not possible or can be reliably and readily detected and suppressed. To achieve this, Section 2.2.4 of Chapter 4 of the passive Requirements Document requires the reactor core to be designed to ensure that fuel safety limits will not be exceeded for any power oscillations through inherent characteristics or prevention by the reactor protection system. The objective is for the reactor core to be free of power oscillations or be self-damping against power oscillations throughout core life.

This requirement is acceptable. However, the staff is concerned with how this design objective can be demonstrated in light of the large uncertainties that currently exist in predicting the stability of power operations. Tests need to be performed to demonstrate the system design capability and to verify and

validate the evaluation methodologies including uncertainty treatment to ensure that this design objective is met. The staff addresses this issue further under "Stable Operation" in Section 4.2.1 of this chapter.

2.2.5 Margin for Normal Operation and Transients

GDC 10 requires the reactor core and associated coolant, control, and protection systems to be designed with appropriate margin to ensure that SAFDLs are not exceeded during any condition of normal operation or anticipated operational occurrences. Section 2.2.5 of Chapter 4 of the passive Requirements Document specifies that a minimum margin of 15 percent will be provided on the limiting power parameters (as defined in the plant technical specifications) used in evaluating compliance with applicable regulatory fuel design requirements. This 15-percent operational margin refers to a demonstrated margin over and above routine margins for design transients, calculational uncertainties, measurement and setpoint tolerances, and so forth, included in the design. The staff concludes that this requirement exceeds that imposed by GDC 10 and is, therefore, acceptable.

2.2.6 Reactivity Control Reliability

Section 2.2.6.1 of Chapter 4 originally required the reactivity control systems to be designed to ensure that they will perform their safety functions under normal, moderately frequent, infrequent, and limiting fault events and that they will be able to perform their intended functions following an earthquake. In a letter dated April 15, 1991, EPRI responded to a staff question regarding whether the systems are required to operate automatically or to be capable of insertion on manual scram following an earthquake. EPRI stated that the safety functions of the reactivity control systems for postulated events including earthquakes will be defined as part of the design process, and if automatic or manual operation of those systems during or after an earthquake is necessary to meet regulatory requirements, it will be identified as a safety function during the design process. Therefore, the requirement in Section 2.2.6.1 of Chapter 4 regarding the capability of the reactivity control systems following an earthquake was redundant and had been deleted in Revision 2 of the passive Requirements Document. In the DSER for Chapter 4, the staff stated that it requires that the reactivity control systems, being safety systems, be capable of insertion after an earthquake to comply with GDC 2 and 22. Therefore, deleting rod insertion capability after an earthquake was not acceptable and could not be used by the plant designer as guidance. This was identified as an open issue in the DSER. By letter dated September 23, 1992, EPRI stated that the sentence concerning an earthquake was deleted to avoid confusion regarding the scope of Section 2.2.6.1 because the decision as to whether or not to require the control rods to be inserted after an earthquake belonged in the area of plant control (Chapter 10) and not in the area of reactivity control reliability. However, Section 2.2.6.1 of Chapter 4 was revised in Revision 4 by requiring that the reactivity control systems be able to perform their intended functions following an earthquake (safe shutdown earthquake). The staff concludes that this is acceptable; therefore, this open issue is closed.

2.2.7 Shutdown Margin

Section 2.2.7 of Chapter 4 requires the reactivity control systems to be designed so that the reactor can be maintained at least 1 percent subcritical in the cold condition at the most reactive time in its life with the highest worth control rod fully withdrawn from the core. In response to a staff request for additional information dated February 22, 1991, EPRI, by letter dated April 25, 1991, stated that the reactivity requirement in Section 2.2.7 of Chapter 4 referred to the net core reactivity (i.e., margin below criticality), not the required worth of the reactivity control systems. The actual total required worth would be a function of the details of the reactor fuel and fuel cycle design. EPRI stated that a steamline break was probably the most restrictive requirement for accidents and that the total worth would be greater than 1-percent reactivity. Since there is no intent to depart from current NRC-approved practice regarding reactivity shutdown margin, the staff concludes that this clarification is acceptable.

2.2.8 Criticality Margin

Section 2.2.8 of Chapter 4 originally required the reactor core and associated components and systems to be designed so that subcriticality is maintained (i.e., K_{eff} less than 0.95) during fuel handling, inside and outside the reactor core, for any single failure or procedural error. For PWRs, the water may be considered borated to the minimum amount permitted for refueling or spent fuel storage, as appropriate.

The staff position is that credit may be taken for borated water inside the reactor core during fuel handling, but nonborated water must be assumed for fuel handling and storage outside the core, even for PWRs (see Chapter 7 of this report), except during accident conditions when the double contingency principle does not require that two unlikely, concurrent abnormal events be considered. In addition, as discussed in NRC Bulletin 89-03, "Potential Loss of Required Shutdown Margin During Refueling Operations," although analyses are performed for PWRs to confirm that the refueling boron concentration is sufficient to maintain the required 5-percent shutdown margin for the final core configuration, these analyses may not be sufficient to ensure that the shutdown margin will be maintained for all intermediate fuel assembly positions. Therefore, any intermediate fuel assembly configuration intended to be used during refueling must be identified and evaluated to ensure that sufficient refueling boron concentration exists to maintain the 5-percent shutdown margin. To address the staff's concern, EPRI modified Section 2.2.8 of Chapter 4 in Revision 2 of the passive Requirements Document by requiring that subcriticality also be maintained under planned intermediate fuel configurations as specified by NRC Bulletin 89-03. The staff concludes that this is acceptable. However, each ALWR designer must also ensure that nonborated water is assumed in evaluating fuel handling and storage outside the core, except when the double contingency principle applies.

2.3 Equipment Design Requirements

2.3.1 Materials

Section 2.3.1.7 of Chapter 4 of the passive Requirements Document requires that low-temperature overpressure protection (LTOP) be provided for the PWR as

specified in Chapter 3 for the reactor shutdown cooling system (RSDC). The staff's evaluation of LTOP is provided in Section 3.3 of Chapter 3 of this report.

Section 2.3.1 of Chapter 4 states that since the reactor system components are critical components, as defined in Chapter 1, vendors are to identify to the plant owner in writing all the materials used in the reactor systems. They also must identify any materials that are not proven as defined in Chapter 1 and obtain approval from the plant owner before they are used. The reactor system materials must comply with the applicable regulatory requirements and American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) material specifications.

Section 2.3.1 of Chapter 4 requires that the RPV base material be SA-508, Class 2 or 3, or SA-533, Grade B, Class 1, and satisfy Appendix G to 10 CFR Part 50. Welding material for the RPV will be of an equivalent composition and satisfy ASME SFA-5.23. RPV cladding material will be Inconel 82 SFA 5.14 Type ER NiCr-3 or Type 308L stainless steel weld metal, except that Type 309L stainless steel weld metal will be used for the first layer of stainless steel cladding.

EPRI requires that the RPV base material and welding material chemistry be controlled so that susceptibility of the material to neutron embrittlement damage is as low as practicable. As a minimum, the requirements given in Section 2.3.1.6 of Chapter 4 must be met. In addition to the chemistry requirements of the applicable ASME Code material specifications, the following chemistry requirements apply for material in or adjacent to the reactor beltline (i.e., where neutron fluence is more than 1.0E+17 neutrons/cm^2):

Base Material		Welding Material	
Phosphorous	0.012% maximum	Phosphorous	0.012% maximum
Copper	0.03% maximum (PWR)	Vanadium	0.05% maximum
	0.05% maximum (BWR)	Copper	0.08% maximum
Sulfur	0.015% maximum	Sulfur	0.015% maximum

EPRI states that the fracture toughness requirements in ASME Code, Section III, Subsection NB-2300, apply. In addition, for material for which impact tests are required, fracture toughness (K_{IC}) will be established as a function of temperature over the range of nil-ductility transition temperature (RT_{NDT}) to RT_{NDT} + 60 °F. The J_{IC} method as specified in American Society for Testing and Materials (ASTM) E-813 may be used.

The following guidelines apply when using stainless steel welding materials with a high nickel content:

- Stainless steel welding material chemistry will be such that sensitization is minimized. The average equivalent delta ferrite must be no less than 8FN (ferrite number) with no reading below 5FN.

- Material processing and vessel fabrication will minimize material sensitization and will not adversely affect resistance of the material to

intergranular stress corrosion cracking (IGSCC). As a minimum, NRC Regulatory Guide (RG) 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components," will be followed.

- Use of Inconel 182 filler metal is prohibited in welds exposed to primary coolant.

Section 2.3.1.4 of Chapter 4 requires that the RPV closure stud and nut material conform to the guidelines in NRC RG 1.65, "Materials and Inspections for Reactor Vessel Closure Studs."

EPRI states that materials and fabrication procedures will be chosen so that furnace and weld sensitization, irradiation-assisted stress corrosion cracking, and crevice corrosion cracking do not lead to IGSCC problems that limit the life of the reactor vessel and those portions of the internal components that are not designed to be replaceable in less than 60 years. Materials that have been shown to be susceptible to IGSCC will not be used. Except for the core belt region in the PWR, the RPV must be designed and fabricated so that the initial RT_{NDT} at the most limiting location is not greater than 10 °F. The initial RT_{NDT} in the PWR core belt region must not exceed -20 °F. The calculated end-of-life (60 years of service) shift in RT_{NDT} caused by irradiation for core belt materials (calculated delta RT_{NDT} as specified in RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials") must not exceed 30 °F for both the PWR and BWR.

A material surveillance program to monitor vessel irradiation and its effect on reactor vessel material properties is required. The surveillance program must comply with 10 CFR Part 50, Appendix H, and ASTM E-185-82.

Surveillance capsules will be inserted in the reactor vessel at the beginning of plant operation. The surveillance program will provide for replacement surveillance capsules and a means for their installation. Fracture toughness specimens must be of sufficient size that valid measurements can be made, and the samples will include sufficient spare material to fabricate additional test specimens. Surveillance requirements in 10 CFR Part 50, Appendix H, and ASTM E-185-82 apply. For each reactor vessel beltline forging exposed to end-of-life fluence in excess of 1.0E+18 neutrons/cm^2, additional surveillance capsule specimens will be included for determining upper-shelf K_{IC} at RT_{NDT} +60 °F. The J_{IC} method as specified in ASTM E-813 will be used to determine K_{IC}. The surveillance capsule will be located near the vessel wall at a position where the exposure lead factor is in the range of 1.2 to 1.5.

The surveillance program must accurately monitor the irradiation environment of the reactor vessel beltline material over the entire vessel design lifetime considering both the magnitude of fluence experienced and changes in fuel management.

The staff concludes that these requirements are acceptable because the reactor and reactor system components will be designed and constructed to achieve a high level of reliability. The RPV material and welding materials are required to meet strict limits on phosphorous, copper, and sulfur, which will reduce the susceptibility of the material to neutron damage. RPV materials are also required to have adequate toughness to meet the requirements of the current ASME Code and NRC regulations. The RPV surveillance capsules program

must meet the requirements of 10 CFR Part 50, Appendix H, and ASTM E-185-82. Fabrication of stainless steel and of welding materials with a high nickel content will be controlled to minimize sensitization and IGSCC.

2.3.2 Vessel and Internals Design

Vessel and Internals Design Life

Section 2.3.2.1 of Chapter 4 requires that the RPV and nonremovable internals, RPV nozzles, and safe ends have a design life of 60 years. The plant designer will establish irradiation exposure or stress and strain limits or both for the RPV internals (including bolting) based on applicable data from operating nuclear plants or materials test reactors or both. The reactor vessel pressure boundary material must meet 10 CFR Part 50, Appendix G, throughout the design life.

In the DSER for Chapter 4, the staff stated that this requirement was acceptable. However, the staff concluded that features should be incorporated into the internals design to allow inservice inspection to detect severe degradation or failure or both. Internals not accessible without major reactor disassembly should be inspectable in place. The staff identified inservice inspections for reactor internals as an open issue. In a letter dated September 29, 1992, EPRI revised the passive Requirements Document to require that the reactor design provide for inservice inspection of reactor internals. Those internals permanently installed in the reactor vessel must now be inspected in place. The staff concludes that this is acceptable and this DSER open issue is closed.

Section 2.3.2.1.5 was added to Chapter 4 in Revision 2 of the passive Requirements Document to require that the RPV internals be designed so that flow-induced vibration will not cause unacceptable damage to the fuel assemblies or the reactor vessel internal structures. Table B.1-2 in Appendix B to Chapter 1 contains an unconditional commitment to RG 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing." For reactor internals identified as a prototype as defined in RG 1.20, the following programs will be completed in accordance with a schedule that is also defined in the guide:

- a vibration analysis program during the design phase
- a vibration measurement program during the preoperational startup testing phase
- an inspection program following the preoperational startup testing

The results of all of the above programs are to be documented.

For reactor internals identified as non-prototype, the preoperational test programs will be as defined in RG 1.20 for the applicable category of non-prototype configuration.

On the basis of the requirements in Section 2.3.2.1.5 of Chapter 4 and the commitment to RG 1.20 in Table B.1-2 of Chapter 1, the staff concludes that the combination of predictive analyses, tests, and post-test inspection

provides adequate assurance that the ALWR reactor internals will, during their service life, withstand the flow-induced vibrations of the reactor without loss of structural integrity.

Vessel Fabrication

Section 2.3.2.4 of Chapter 4 imposes the following requirements on reactor vessel fabrication:

- The RPV must be fabricated with all welds made and finished in the shop. The location and finish of all welds must be suitable for future automatic inservice inspection (ISI) in accordance with Section XI of the ASME Code.

- The PWR cylindrical vessel sections must be forged shells. The beltline forged sections of the vessel must be of such a length that the active core region is free of welds. The uppermost forged section must be an integral-type design in which the vessel flange and adjacent shell course are formed as a single reinforced ring forging.

- For the BWR, the beltline shell course must be a seamless ring-forged section of such a length that the active core region is free of welds. The lower head and the cylindrical sections must be made from as few forgings as possible.

- The electroslag welding process must not be used for any weld seam of the reactor vessel.

- All RPV cladding must be weld-deposited, and the finish of the cladding must permit ISI without further grinding or finishing.

- RPV cladding requirements are as follows:

  - For PWRs, clad areas must include all surfaces that are wetted with borated water during operation or refueling. This includes the wetted area of the vessel flange and the area around its stud holes.

  - For BWRs, all RPV interior surfaces must be clad except for the closure head, the nozzles, and the nozzle weld zones.

- Pressure boundary parts made of stainless steel or NiCrFe Alloy 600 (except for cladding and nozzle buttering) must not be furnace sensitized. Nozzle buttering, which will be furnace sensitized, must be limited to high-chromium, low-carbon grades of NiCrFe, such as Inco 82.

The staff concludes that these requirements are acceptable. The required fabrication control will ensure that the RPV will perform in service as designed and that all finished welds will be suitable for automatic inservice inspection.

Vessel Inservice Inspections

Section 2.3.2.6 of Chapter 4 requires that access and equipment be provided so that automated ISIs are possible for all RPV welds (including nozzle welds) to reduce inspection time and personnel exposure. The staff concludes that this requirement is acceptable.

Refueling Cavity Seal

Section 2.3.2.7 of Chapter 4 requires that the ALWR have a permanent seal between the RPV and the surrounding refueling canal floor to permit flooding above the vessel during refueling. The refueling cavity seal must meet structural design requirements consistent with seismic Category I requirements, must be designed to maintain seal integrity within a specified maximum permissible leak rate for a service lifetime of 60 years, and must be designed for testing before flooding. Because use of a permanent seal eliminates the critical path time, and the resulting personnel radiation exposure, needed to install and remove a nonpermanent pool seal for each refueling outage, the staff concludes that this is acceptable.

Reactor Bolting

Section 2.3.2.8 of Chapter 4 requires that the use of high-strength bolting materials (which have been known to degrade while in service) be minimized. The design of the RPV flange bolting, RPV support bolting, and RPV internal bolting important to safety must incorporate features to allow inservice inspection to detect severe degradation or failure or both. Bolting not accessible without major reactor disassembly must be inspectable in place. Because some thread lubricants have caused corrosion and cracking of bolting, only thread lubricants demonstrated to be effective may be used. Leak sealants will be prohibited for reactor systems because they contain materials that are corrosive. The staff concludes that these requirements are acceptable because they will serve to minimize degradation and will ensure that reactor bolting failures do not occur.

Insulation

Section 2.3.2.9 of Chapter 4 requires that the RPV insulation be of the standard stainless steel reflective type (which has proven to be compatible with the intended service). The insulation must have an integral air gap, be capable of draining condensation on the inner surfaces, and be easily removed for maintenance or vessel cleanup in the event of an inadvertent actuation of the depressurization system. It must be configured so that an annular space exists between the RPV and the insulation (i.e., standoff-type insulation) so that inservice inspections of the vessel can be performed without removing the insulation. The staff concludes that these requirements specify features that minimize heat losses from the RPV, minimize heat load on containment cooling systems, incorporate provisions to facilitate maintenance and reduce personnel exposure and are, therefore, acceptable.

2.3.3 Core and Fuel Design

Fuel Assembly Reconstitutability

Section 2.3.3.2 of Chapter 4 requires the fuel assemblies to be designed so that all fuel rods can be reconstituted. Reconstitution includes the partial disassembly of irradiated fuel assemblies for inspection, repair, or other reasons, and the subsequent reassembly. The staff concludes that this requirement is acceptable provided the full assembly reconstitution conforms to the guidance of Generic Letter (GL) 92-02, "Alternative Requirements for Fuel Assemblies in the Design Features Section of Technical Specifications," issued on July 31, 1992. The staff will review applications for FDA/DC to ensure that the following are properly addressed:

- use of NRC-approved methodologies for safety evaluation that are applicable to the scope and configurations of the reconstituted fuel assemblies, such as "dummy rods" and "vacant water space" configurations

- proper consideration of the effects on mechanical designs, such as the effects of differential thermal expansion on the proper seating of the fuel rod or on the relaxation of the spacer spring that could lead to fretting wear

- analysis of the changes in fuel designs to confirm that the seismic and loss-of-coolant-accident design loading conditions will not cause any structural deformation that could result in the loss of fuel coolable geometry or prevent control rod insertion, although these changes in the fuel designs could affect the grid strength or the mass, stiffness, and fundamental frequency of the fuel assembly

3 BWR REACTOR PRESSURE VESSEL AND INTERNALS

3.1 Scope

The RPV is the major element of the reactor coolant system pressure boundary and will contain and support the reactor core and reactor internals. The reactor coolant supply will circulate through the RPV and maintain a floodable volume to keep the core covered. The internals will provide the supporting elements and devices inside the RPV that, together with the reactor core, reactor coolant system, and RPV instrumentation, will perform the function of nuclear steam generation.

Section 3 of Chapter 4 of the passive Requirements Document, together with the applicable portion of Section 2 of Chapter 4, provides the utility requirements for the RPV and its internals and instrumentation for the BWR plants. The staff concludes that none of them conflict with NRC requirements. However, several items are discussed further below.

3.2 Performance Requirements

RPV Internals

Reduced subcooling of core coolant affects core performance, plant efficiency, and the contribution of natural circulation to the coolant recirculation rate. To minimize the reduction of subcooling of the feedwater/recirculating coolant mixture to the core that results from carryunder condensation, Section 3.2.2.2 of Chapter 4 of the passive Requirements Document originally required the carryunder of steam in the RPV downcomer water to be within an acceptable limit at rated reactor output at the normal water level. In EPRI's letter dated May 8, 1991, responding to a staff question dated March 8, 1991, concerning a quantitative acceptable limit, EPRI stated that the carryunder of steam in the downcomer water was normally set at about 1 percent at rated reactor output and at normal water level, and that it was a detailed design parameter that was established by a combination of analysis and experience, and that the exact value, which is a function of the detailed design of the core and fuel, was not a dominant parameter in core performance. In Revision 2 to the passive Requirements Document, the "acceptable limit" in this section was replaced by "1%." The staff concludes that this does not conflict with any NRC regulations. However, since the exact value is a function of the detailed design of the core and fuel, the staff will ensure that the carryunder limit is acceptable during its review of an application for FDA/DC.

Sections 3.2.2.4 and 3.2.2.5 of Chapter 4 require that the RPV internals design configuration, including chimneys at the reactor core exit, provide for a full natural circulation rate that will be stable, with safety margin, in all regions of normal operation. The objectives of these requirements to use chimneys to enhance natural circulation to provide driving head through the core and provide margin to accommodate transient conditions are acceptable. However, the passive Requirements Document does not specify test requirements to demonstrate the capability of first-of-a-kind system designs and to verify and validate the codes and methodologies used for safety analyses with respect to the system designs. The staff will need to evaluate test requirements and

verification and validation of codes and methodologies during its review of an application for FDA/DC. These items are also discussed under "Stable Operation" in Section 4.2.1 of this chapter.

Section 3.2.2.8 of Chapter 4 requires that highly irradiated (e.g., neutron fluence greater than 2.5E+21 neutrons/cm^2 at energy greater than 1 megaelctronvolt (MeV) RPV stainless steel internals be constructed from materials that are resistant to radiation-induced stress corrosion cracking unless alternative materials can be justified. Special consideration will be given to the design and construction of internals that will experience neutron fluences of 5E+20 to 2.5E+21 neutrons/cm^2 at energy greater than 1 MeV. Limits on sustained stress, induced strain, and special alloys resistant to irradiation-assisted stress corrosion cracking will be used. The staff concludes that these requirements are acceptable because they will provide long-term protection against stress corrosion cracking.

3.3 Equipment Design Requirements

Reactor Pressure Vessel

- RPV Support

  Section 3.3.1.1 of Chapter 4 of the passive Requirements Document requires that the RPV support be welded to the cylindrical wall of the vessel below the bottom of the core and be bolted to the reactor pedestal. The support will not interfere with installation or removal and maintenance of the equipment extending from the reactor vessel or connections to this equipment. This support will resist compressive and tensile loads and will accommodate thermal expansion by flexure.

  The staff concludes that these requirements are acceptable because the RPV support will be fastened to the vessel at a point away from the core to keep the fluence low at the junction.

- RPV Material Temperature Instrumentation

  Section 3.3.3.3 of Chapter 4 specifies that temperature measurement instrumentation will be provided only as necessary to implement plant operating procedures (e.g., to monitor metal temperature differentials during plant startup and shutdown) and to ensure compliance with technical specifications and plant operating limits.

  The rationale portion of Section 3.3.3.3 originally stated, "With the improved $(RT)_{NDT}$ of the ALWR, thermocouples would not be required for brittle fracture protection of the RPV." In the DSER for Chapter 4, the staff stated that this statement was confusing because EPRI had specified the use of thermocouples in the RPV and that EPRI must delete the sentence. This was identified as an open issue. In Revision 4, EPRI revised the passive Requirements Document by deleting this sentence. The staff concludes that the ambiguity has been removed from Section 3.3.3.3. Therefore, this open issue is closed.

4 BWR CORE AND FUEL

4.1 Scope

The core and fuel are required to generate heat to produce steam for electrical power generation while being controlled to operate within nuclear, thermal, and mechanical limits. Nuclear reactivity will be controlled by movable poison control elements, supplementary burnable poison material, and injection of soluble poison under emergency conditions. These functional requirements are no different from those for the current generation of BWRs.

4.2 Performance Requirements

4.2.1 Core and Fuel Power Generation

Stable Operation

Section 4.2.1.2.1 of Chapter 4 of the passive Requirements Document requires that the core characteristics allow stable operation, which is defined as core-wide and channel decay ratios that are to be within 0.4 and 0.3, respectively, for all expected steady-state operating conditions and moderate-frequency transients. The rationale portion of this section also states that analysis methods and procedures should be consistent with current techniques accepted by the NRC. Section 4.2.1.2.2 of Chapter 4 requires that instrumentation be provided to detect core-wide and/or regional instabilities and that the instrumentation include a display and alarm to enable the control room operators to recognize an instability before the fuel safety limits are exceeded. This requirement will enable detection of oscillations so that the reactor can be placed in a stable state before safety limits are exceeded.

Though the core-wide and channel decay ratio limits of 0.4 and 0.3 appear to provide ample margins to account for uncertainties in predicting the stability of power reactors, further extensive stability studies will be needed to evaluate stability under transient and anticipated transient without scram (ATWS) conditions. The staff performed a preliminary assessment of the thermal-hydraulic stability characteristics of the simplified boiling water reactor (SBWR) design having a single open chimney. The assessment was based on preliminary design information provided by General Electric (GE). The preliminary results of the assessment using the LAPUR frequency domain stability code showed that the system appears to be very stable under normal operating conditions. However, abnormal conditions of operation might be reached under credible transient sequences that could result in the onset of density-wave power and flow oscillations. In addition to the density-wave instability, natural circulation reactors are known to be susceptible to other kinds of instabilities, such as (1) geysering instability that may occur between parallel channels at very low flow during startup and may result in large flow oscillations leading to dryout conditions and (2) the loop or enthalpy-wave instability due to the long riser delay that results in low-frequency loop-flow oscillations.

The staff was also concerned that the passive BWR with a long, single open chimney, being of new design, might be susceptible to unknown types of instabilities that the normal stability codes are not able to predict. Of

particular concern was the lack of experimental data that are applicable to the single open chimney with the outlet pressure drop induced by the separators. In response to this concern, GE indicated that the SBWR design had been changed from a single-open-chimney to a divided-chimney design. GE claimed that existing experiments were representative of the divided-chimney design now used and will submit descriptions of these experiments for staff review. If the staff agrees that these experiments are suitable for code validation, GE will validate its codes against these experiments and provide the results for staff review.

The staff concludes that the requirements of Sections 4.2.1.2.1 and 4.2.1.2.2 of Chapter 4 and the performance requirements of Sections 2.2.4, 3.2.2.4, and 3.2.2.5 of Chapter 4 properly state the ALWR objectives of providing stable operation of BWRs. However, the passive Requirements Document does not provide a specific BWR design or the appropriate process to implement these requirements. The staff will review the stability issue for a specific BWR, such as the SBWR, design during its review of an application for FDA/DC. The staff has identified the following items as being needed to address the stability issue:

- Identification of all types of stability that can occur during normal operational transients and various transient scenarios including ATWS.

- Verification and validation of the computer codes, such as TRACG and FABLE, used for the stability analyses to ensure their capability to analyze all types of instabilities identified.

- Availability of appropriate experimental data or tests to verify and validate analysis codes and to confirm that the passive BWR designs are stable under conditions of all ranges of operation and transients. The test facilities must be comparable to the specific BWR designs, including configuration and characteristics of important systems and components and overall loop configuration. The test procedures must be thoroughly considered, and the test conditions must encompass ranges of operating conditions including various transient scenarios. The scaling effects must be properly considered.

- Plant-specific analysis with the NRC-approved codes and methodologies to confirm that the core-wide and channel decay ratio limits of 0.4 and 0.3, respectively, provide sufficient margin to ensure stable operation during design-basis transient and accident conditions.

In addition, the decay ratios are dependent on the calculational methodologies, including the codes and uncertainty treatments. In a letter dated March 8, 1991, the staff commented that the specific values of the decay ratio limits to ensure stability should be tied to the specific methodologies used for the decay ratio calculations. In a letter dated May 8, 1991, EPRI responded that the limits specified in the passive Requirements Document were based on the methods and procedures being made final by the BWR Owners Group (i.e., FABLE frequency domain calculations, normal and transient events, more conservative inputs, etc.). The staff concludes that this clarification is acceptable. However, it requires that the decay ratio limits and the analysis methods and procedures be provided as part of an individual application for FDA/DC.

Core and Fuel Design Bases

Sections 4.2.1.3.3 and 7.2.1.3.1 of Chapter 4 originally required the fuel design to meet the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K, for a postulated loss-of-coolant accident (LOCA). These requirements precluded the use of realistic, best-estimate (BE) evaluation methods under the new emergency core cooling system (ECCS) rule of 10 CFR 50.46. In response to a staff letter dated March 8, 1991, EPRI stated in a letter dated May 9, 1991, that it did not intend to preclude the use of realistic, BE evaluation methods for the passive plant. These requirements were revised in Revision 2 of the passive Requirements Document by deleting the reference to 10 CFR Part 50, Appendix K. EPRI also stated that the plant designer would make the choice of using either the BE or the Appendix K approach in the LOCA analysis.

Even though the new ECCS rule allows the use of either BE or Appendix K approaches, the staff concludes that BE evaluation methods should be used for LOCA analyses of passive plants because Appendix K was originally written for the current generation of LWRs where ECCS injection arrangements are different from those in the passive plant designs. Many Appendix K features might not be applicable to the passive ALWR designs and would require additional design and staff review to justify their use. In addition, a BE calculation would provide a more realistic evaluation of plant response to a LOCA event and therefore provide better information for developing emergency procedures.

Load-Following and Maneuvering Capability

In requiring that the core design include load-following and maneuvering capability, Section 4.2.1.4.2 of Chapter 4 specifies that no preconditioning will be required of the fuel for maneuvering. The passive Requirements Document states that the intent of this requirement is to remove previous limiting fuel characteristics, such as pellet-cladding interaction, for plant maneuvering so that components and systems other than the fuel establish the plant maneuvering limits. It further indicates that improved designs of fuel and reactivity control systems that meet this requirement have been demonstrated. The staff recognizes that this is a desirable objective; however, the passive Requirements Document does not specify the required analyses to be performed to demonstrate independence of plant maneuvering limits from fuel conditions. In response to a staff request for additional information dated March 8, 1991, EPRI stated in a letter dated May 8, 1991, that the specific analyses needed to ensure meeting this objective depend on the details of fuel designs and fuel management schemes adopted, and that it is expected that the plant designer would include mitigating design features (e.g., fine-motion CRD and barrier fuel) as part of his response to achieve this objective. The staff will evaluate load-following and maneuvering capability during its review of an application for FDA/DC.

Nuclear and Thermal-Hydraulic Design

Section 4.2.1.6.2 of Chapter 4 requires that the BWR core design provide for extended cycle operation at reduced power or with reduced feedwater temperature. Since operation with reduced feedwater temperature could result in less thermal-hydraulic stability, an analysis must be done to determine the benefits of extending cycle operation against the undesirability of decreasing

core stability by reducing feedwater temperature. The staff transmitted this concern in a letter dated March 8, 1991. In response to this concern, in a letter dated May 8, 1991, EPRI stated that the designer will consider the possibility of reducing feedwater temperature to extend cycle length in the SBWR stability calculation. Analysis procedures will be consistent with those recently developed by the BWR Owners Group and used at operating plants. The decay ratios for the case with end-of-cycle reduction in feedwater temperature will be calculated and be subject to the stability limits specified by EPRI.

GE Topical Report NEDO-31960, "BWR Owners Group Long-Term Stability Solutions Licensing Methodology," describes the methodology developed by the BWR Owners Group to meet the BWR stability objective by either preventing stability-related neutron flux oscillation or detecting and suppressing the oscillation before the specified acceptable fuel design limits are exceeded. The staff has reviewed this topical report and finds it acceptable for current BWR designs and the ABWR. If the BWR designer uses this methodology for the passive ALWR (i.e., SBWR) stability analysis, it should ensure that the methodology is applicable to the passive ALWR design. In addition, the end-of-cycle reduction in feedwater temperature should be analyzed and reviewed during plant-specific design. The staff will review the specific use of this methodology, or any other proposal, during its review of an application for FDA/DC.

4.2.2 Fuel Reliability, Burnup and Lifetime

In regard to fuel reliability, Section 4.2.2.1 of Chapter 4 requires that premature fuel failure due to manufacturing defects be less than 1 per 50,000 fuel rods with a goal of 1 per 100,000. EPRI also states that recent industry experience has shown that 1 failure per 50,000 rods is an achievable reliability. Section 4.2.2.2 requires fuel mechanical design to be capable of peak bundle-average burnup of at least 50,000 megawatt-days per metric ton uranium (MWD/MTU). This peak burnup requirement of 50,000 MWD/MTU is inconsistent with the 45,000 MWD/MTU requirement of Section 3.6 of Chapter 1. However, either of these values is higher than the current burnup limit of approximately 44,000 MWD/MTU for existing BWR fuel. The staff requires, to support this high fuel burnup operation, sufficient data on high fuel burnup to demonstrate fuel integrity in the areas of fission gas release, cladding corrosion due to oxidation and hydriding, and reduction of cladding material strength. EPRI states that the intent of this requirement is to define minimum acceptable burnup values consistent with present or expected near-term experience. These requirements are not based on any specific database extending to the specified burnup level. The burnup limit actually placed on ALWR fuel would be that demonstrated and licensable by a specific fuel vendor at the time of use. This issue is also identified in Section 3.6 of Chapter 1 of this report.

4.2.3 Structural Design Bases - Fuel Assembly

Section 4.2.3 of Chapter 4 requires that fuel assemblies, including the channel, be designed for the load combinations and loading conditions defined in Section 4 of Chapter 1 to address peak fuel design loading that must be accommodated. Under those conditions, the fuel must not prevent insertion of control rods or damage any component of the fuel assembly in such a way that

would prevent safe handling. Because these requirements will enhance hydrodynamic stability and ensure that these components perform as designed, the staff concludes that this is acceptable.

4.2.4 Fuel Channel Lifetime

Section 4.2.4 of Chapter 4 requires the BWR fuel channel to be designed for a lifetime of two fuel-bundle in-core residence times with the objective of reusing channels on more than one fuel assembly. The staff addressed its concern regarding the channel box bow effect on the critical power ratio for channel boxes in the second bundle lifetime operation in NRC Bulletin 90-02, "Loss of Thermal Margin Caused by Channel Box Bow."

In EPRI's letter dated May 8, 1991, responding to a staff request for additional information dated March 8, 1991, EPRI stated that a two-cycle channel lifetime was an objective, not a requirement. The effects of channel bowing, as well as other irradiation effects such as corrosion and irradiation creep, must be properly accounted for so that core operation remains within its licensed limits. The two-cycle channel lifetime has been demonstrated as feasible for selected channels built to suitably stringent requirements and operated under a fuel management procedure that accounts for channel irradiation effects. Individual utilities will decide whether to use the two-cycle channel lifetime at the time of use, and the staff will evaluate this use at that time.

The staff concludes that EPRI has properly addressed its concern. If the two-cycle channel box is used, the staff will evaluate channel bowing and other irradiation effects. However, the applicant for FDA/DC will have to demonstrate the acceptability of a two-cycle channel lifetime, if used.

4.2.5 Core Nuclear Reactivity Control

Section 4.2.5 of Chapter 4 requires that the core and core controls meet the requirements of Sections 2 and 4.2.1 of Chapter 4 of the passive Requirements Document as basic requirements for safety, plant protection, and length of operating cycle. In addition, EPRI requires that the core-wide coolant void coefficient be negative for all operating conditions. The negative core void coefficient will alleviate concerns about potential positive feedback mechanisms that could cause accidents. The staff concludes that this is acceptable.

4.2.6 Control Rods

Section 4.2.6.2 of Chapter 4 requires that the BWR control blades used for maximum core insertion be designed with minimum exposure capability of 4E+21 neutrons/cm^2, with a target of 8E+21 neutrons/cm^2, and that the blades not used for maximum core insertion be designed for an operating life of 13 or 20 reactor full-power years. This requirement is beyond the operating experience data of the current LWRs. To support the desired extended operating life of the control blades and control rod assemblies, sufficient performance data will be required to demonstrate structural integrity under irradiation effects, including material hardening, absorber depletion, and swelling. This issue is also identified in Section 3.6 of Chapter 1 of this report.

4.3 Equipment Design Requirements

Section 4.3.2.2.2 of Chapter 4 of the passive Requirements Document requires that core power distribution be monitored both by fixed power level in-core sensors and by movable traversing in-core probe (TIP) detectors or fixed incore calibration detectors. If the TIP system is used, the TIP drive mechanisms, guide tubes, and position indicators, including any necessary shielding and TIP motion interlocks, will be designed and located for ease of servicing or replacement. The TIP system has been the source of several overexposures or near overexposures (from exposure to irradiated in-core detectors or the attached TIP drive cables) in recent years during TIP withdrawal or replacement operations. The TIP interlock design will include appropriate alarm warnings and protective measures. Because these features reduce personnel doses associated with TIP operations, the staff concludes that they are acceptable.

5 BWR CONTROL ROD DRIVE SYSTEM

5.1 Definition and Scope

The control rod drive (CRD) system will include the electrohydraulic CRD, rod drive motors, hydraulic control units, hydraulic supply system, scram and scram pilot solenoid valves, air header dump valves, interconnecting piping, and associated instrumentation including rod position and separation sensors. The BWR CRD system is required to perform the following functions:

- withdraw and insert the control rods at a normal rate for operational control
- control and indicate the positions of the control rods throughout the full stroke
- insert the control rods for shutdown (scram) at the high rate required to maintain fuel integrity
- control the position of ganged-rod groups for faster changes in rod position
- provide for insertion of control rods by an alternative and diverse method on receipt of anticipated transient without scram (ATWS) signals
- provide reactor coolant makeup water

5.2 Performance Requirements

Section 5.2 of Chapter 4 of the passive Requirements Document establishes the performance requirements for the CRD system.

Reactivity Control

Section 5.2.1.2 of Chapter 4 requires that limiting conditions for operation be developed to define the acceptable number and arrangement of CRDs that are found to exceed the maximum scram times during test or operation. In the rationale portion of this section, EPRI recognizes that, regarding the use of such CRDs, a more precise calculation may be carried out by considering the actual performance of nearby control rods, measurement errors, and the current core operating conditions. EPRI anticipates that the use of CRDs with scram times in excess of the maximum may help to meet the plant availability goals (e.g., annual average of 87 percent over the life of the plant) without reducing safety. The staff concludes that these requirements do not conflict with NRC regulations.

Section 5.2.1.4 of Chapter 4 requires that the scram performance and design of the CRD and hydraulic system accommodate an all-hafnium type, a boron carbide type, or other types of proven control rods in BWRs. The scram performances of the different blades are not required to be the same. The hafnium-type control rods have slower scram times because they are heavier. This requirement is acceptable provided the scram time and performance are shown to be satisfactory in the plant-specific safety analysis.

Section 5.3.1 of Chapter 4 requires that the withdrawal and insert motion at normal speed be provided by electric motor drives and that the scram action of the CRD be achieved by water hydraulic pressure taken from gas-charged accumulators. This arrangement would allow each accumulator to provide scram pressure for several CRDs if the concept were adequately supported by a safety evaluation. EPRI states that these requirements are based on favorable experience with the electric motor drive design in BWRs overseas. The capability to move rods in small increments permits more precise core power shaping and reduces the tendency for cracking of the fuel cladding associated with large increments. In addition, the use of electric motor drives enables simpler seals to be used and allows changing these without removing the CRDs. Sections 5.2.1.5 and 5.2.1.6 of Chapter 4 require that the normal withdraw and insertion motion speeds, increments, and repeatability provided by the electric motor drives meet the requirements for diverse reactivity control and plant load control, and a scram signal to initiate rod insertion by both the hydraulic and electric motor drive mechanisms. Section 5.3.6 of Chapter 4 requires the CRD system to include provisions for an alternative means of rod insertion operation using a separate and diverse means (i.e., different from normal scram by the reactor trip system).

The staff concludes that these requirements are consistent with the NRC requirements in 10 CFR 50.62 to provide for an alternative means of rod insertion using a separate and diverse means. Such alternative rod insertion would include diverse actuation logic and an electric motor.

Section 5.2.1.8 of Chapter 4 requires the plant designer to establish, in conjunction with the performance requirements of the core and fuel, normal withdrawal and insertion speeds for the electric drive and scram time. Since the electric motor drive is intended to provide a backup to the hydraulic scram for protection against these events, its scram time must be evaluated for these events. In a letter dated May 8, 1991, EPRI stated that the electric motor-driven train is a single-speed train and that the allowable scram time must be closely coordinated with the core and fuel design. The staff is concerned about possible adverse power peaking and flux tilt at the top of the core caused by slow control rod insertion by the electric motor drive scram during ATWS events. EPRI states that it is not likely that the insertion of all rods at the maximum rate of the motor drive would result in severe power peaking tilt at the top of the core because the time for the control rods to reach one-quarter insertion, which will take the power to virtual zero, will be quite short. However, evaluation performed by GE for the evolutionary advanced boiling water reactor did show the potential for some adverse axial power peaking before the design was modified. Though the staff recognizes that the standby liquid control system has been changed from manual to automatic actuation for the passive BWR designs, as discussed in Section 4.5 of Chapter 5 of this report, the staff will have to evaluate this axial power tilt issue with regard to ATWS mitigation during its review of an application for FDA/DC.

Safety and Reliability

Section 5.2.3.5 of Chapter 4 requires that the CRD design provide a means to prevent or limit the rate of control rod ejection from the core due to a break in the CRD pressure boundary. Section 5.2.3.6 of Chapter 4 requires that the residual torque of the CRD motors prevent the control rod from inadvertently

withdrawing or drifting outward during any nonaccident, accident, post-accident, and seismic condition. Section 5.2.3.7 of Chapter 4 requires that the control rod and CRD drive rod be prevented by redundant means from moving out of the fully inserted position following a scram until the drive rod is reconnected to the motor drive. These design objectives are aimed at preventing a control rod malfunction accident that results in rapid reactivity insertion and fuel damage. In Chapter 1, Table 1.2-1, "ALWR Event Initiators and Frequency Categorization," EPRI uses these design objectives to preclude the control rod assembly (CRA) malfunctions in the passive BWRs from being considered for safety analysis. Since the design details are not available in the passive Requirements Document, if CRA malfunctions are precluded from BWR accident analyses, the staff will perform an evaluation to determine if this is acceptable during its review of an application for FDA/DC.

Section 5.2.3.8 of Chapter 4 requires the CRD system to provide high-pressure reactor coolant makeup capacity. This capacity can be provided by two of the system pumps operated in parallel if another system with the same capacity is also provided to meet the required redundancy, or each of the two CRD pumps may be provided with full capacity for this function. This reactor coolant makeup requirement is consistent with Section 2.3 of Chapter 3, which requires that the CRD system provide the defense-in-depth function of reactor coolant makeup. The staff's position regarding the reliability and availability of the defense-in-depth active systems is discussed in Section 2.3 of Chapter 1 of this report.

Conclusion

The staff concludes that the requirements in Section 5.2 of Chapter 4 of the passive Requirements Document, with the possible exception of the vendor- or utility-specific item concerning the potential axial power tilt at the top of the core due to slow motion of the backup scram protection noted above, do not conflict with NRC regulations.

5.3 Equipment Design Requirements

Section 5.3 of Chapter 4 of the passive Requirements Document describes the equipment design requirements for the CRD system.

Materials

Section 5.3.3 of Chapter 4 requires that materials used in the construction of the CRDs be selected on the basis of the requirements of Chapter 1, taking into consideration suitability for service, the requirements of applicable codes and standards, and a history of satisfactory service in LWR applications.

The staff has reviewed the materials requirements for BWR CRD systems, as discussed in Section 5 of Chapter 1 of the passive Requirements Document, to ensure that materials in the reactor coolant system and related systems are compatible with their intended service and concludes that the requirements are acceptable.

Valves

Section 5.3.5.3 of Chapter 4 requires that the scram pilot solenoid valves be designed for continuous operation at the minimum and maximum voltages and frequencies required by the reactor protection system (RPS) and indicates that the purpose of this requirement is to avoid the overheating and consequent damage to valve materials observed in earlier plants. The staff concludes that this position is acceptable. However, there is no indication as to whether this requirement is meant to replace the requirement for the electric protective assemblies (EPAs) that were used in past BWR designs to prevent overvoltage, undervoltage, or underfrequency conditions from failing the scram pilot solenoid valves in a non-fail-safe state. In its letter dated September 29, 1992, EPRI stated that the requirement of Section 5.3.5.3 was not meant to replace any requirement for EPAs. It was included in the passive Requirements Document to avoid repetition of previous misuses of solenoid valves with voltages or frequencies for which they were not designed. However, EPRI also stated that it considers it inadvisable to use EPAs in the power supplies of the RPS of advanced BWRs. The staff's evaluation of EPRI's position for not requiring EPAs in the RPS power supplies is discussed in detail in Section 7.2.9 of Chapter 11 of this report. The staff concludes that, because the scram ac solenoid valves are the primary means of scramming the reactor and provide immediate insertion of all rods, a system for detecting the occurrence of undervoltage, overvoltage, and underfrequency of power sources to these solenoids should continue to be used in the advanced BWR designs, and that because of the additional diverse means to scram the reactor at advanced plants and the amount of time generally necessary to fail the solenoid valves, an alarm system capable of detecting these occurrences would be acceptable. In Section 7.2.9, the staff also discusses its requirement regarding this alarm system. The staff will pursue the implementation at such an alarm system at the advanced BWRs during its review of an individual application for FDA/DC.

Maintenance

To reduce personnel radiation exposure during maintenance of CRDs, Section 5.3.9 of Chapter 4 requires the use of a machine for automated removal and reinstallation of the CRDs. Adequate working space will be provided below the CRD mechanisms to permit removal of the motors and maintenance on the shaft seals without having to remove the entire mechanism from the reactor. This will reduce both the time needed to perform this work and the resultant exposure incurred. In addition, temporary shielding will be used, if needed, to lower area dose rates during removal and storage of CRD internals.

Conclusion

The staff concludes that the requirements in Section 5.3 of Chapter 4 of the passive Requirements Document do not conflict with NRC regulations and are, therefore, acceptable.

6 PWR REACTOR PRESSURE VESSEL AND INTERNALS

6.1 Definition and Scope

The PWR RPV and its internals will provide a high-integrity pressure boundary containing the reactor coolant, reactor core, and fuel fission products. In addition, the RPV and internals will perform the following functions:

- provide support for fuel assemblies and maintain their orientation and position within the reactor core
- provide the necessary structure that will result in a flow path for the reactor coolant to adequately remove heat generated by the core while ensuring proper reactor flow distribution, resisting upward flow-induced movement of the fuel assemblies, avoiding flow-induced vibration, ensuring positive location and guidance for control rod assemblies, and ensuring that heat generated by each fuel assembly is removed by the reactor coolant
- provide information regarding the RPV water level during shutdown

These functional requirements are no different from those for the current generation of PWRs and are, therefore, acceptable.

6.2 Performance Requirements

Section 6.2.1 of Chapter 4 of the passive Requirements Document requires that the RPV be capable of satisfying all functional requirements under normal, moderate-frequency, infrequent, and limiting fault events. It further requires that a natural circulation cooldown (NCC) transient be evaluated as part of Code vessel evaluations as an infrequent event that could occur at least 30 times over the 60-year life of the RPV. This latter requirement will ensure appropriate analysis is performed to demonstrate the adequacy of the RPV for an NCC of the reactor from full power, and that the design will be able to withstand multiple cooldowns sufficient to resolve Generic Safety Issue (GSI) 79, "Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown." The staff's evaluation of EPRI's requirements to address GSI 79 is presented in Section 3.2.17 of Appendix B to Chapter 1 of this report. The staff concludes that EPRI's requirement to design the plant for at least 30 NCC cycles during its 60-year plant life, consistent with the RCS heat removal capability and the maximum cooldown rate allowed by the plant's technical specifications, is acceptable to resolve the concerns associated with GSI 79.

6.3 Equipment Design Requirements

Section 6.3 of Chapter 4 of the passive Requirements Document specifies equipment design requirements for the RPV, reactor internals, and RPV instrumentation.

RPV Head and Closure Removal

Section 6.3.1.6 of Chapter 4 requires that the ALWR be designed so that remote operation can be used to detension and remove RPV studs and to cover stud holes. In addition, a capability for integrated head disassembly will be provided. This feature will enable the entire head package and all related components to be lifted as a single unit during refuelings. These design features are intended to shorten refueling outages by simplifying several of the refueling outage tasks. Since most of the collective dose at LWRs is incurred during outages, the use of these design features would thereby result in lower personnel exposures.

Refueling Cavity Seal

Section 6.3.1.8 of Chapter 4 requires the PWR refueling cavity seal to be located as high as practicable on the reactor vessel so as to minimize the area of the vessel's exterior material that would be exposed to borated water, which is corrosive to carbon steel. It also requires the seal to be designed so that it is not susceptible to any single failure that could result in a rapid draindown of the refueling cavity. These requirements will minimize the outside area of the vessel that must be protected and ensure that sufficient time will be available to safely secure fuel in the event of the maximum credible seal failure.

Reactor Internals

Section 6.3.2.1 of Chapter 4 requires that the base metal of reactor internals be stainless steel Type 304 or 316. The staff concludes that this requirement is acceptable because these materials have performed well in service.

Thermal Shield

Section 6.3.2.4 of Chapter 4 specifies that thermal shields in the annulus between the core barrel and the RPV will not be used. EPRI states that preliminary evaluations have shown that the reduction in neutron exposure of the RPV resulting from typical thermal shield design is adequately compensated for by other ALWR requirements. The staff is aware that thermal shields in many operating PWRs have been removed. However, specific core and fuel designs must be evaluated to ensure the radiation exposure to the RPV wall will comply with the fracture toughness requirements for protection against pressurized thermal shock specified in 10 CFR 50.61. The staff will evaluate this item during its review of an application for FDA/DC.

Bolting

Section 6.3.2.5 of Chapter 4 requires that the reactor internals bolting be solution-annealed Type 316 stainless steel, with cold working if moderate strength is required. If no alternative to the use of high-strength materials exists, Inconel X-750 may be used, provided the chemistry and fabrication requirements in Section 5 of Chapter 1 are followed. Alloy A-286 material may not be used. If the use of Inconel X-750 cannot be avoided, plant designers must document the basis for the use of high-strength bolting and the controls established to ensure its application.

The NRC staff concludes that these requirements are acceptable because these materials have performed well in service.

RPV Level Instrumentation

Section 6.3.3.1 of Chapter 4 requires that redundant level instrumentation systems be provided for measuring the RPV coolant level during depressurized, shutdown conditions to meet the needs of various maintenance activities. It also requires that the shutdown level monitoring system piping between the reactor coolant system and the pressurizer be permanently mounted, but that it be disconnected by means of blind flanges during normal operation. This requirement for permanent piping is intended to eliminate inaccuracy that has been experienced in current plants with temporarily installed tygon tubes; when the tube kinks, air is entrapped in the tube.

As discussed in Section 9 of Chapter 3 of this report, GL 88-17, "Loss of Decay Heat Removal," specifically requires two independent RCS level measurements during shutdown operation. The staff will review individual applications for FDA/DC to ensure that redundant and independent level instrumentation systems are provided in the design.

The passive Requirements Document originally did not specify a requirement for a reactor vessel level instrumentation system (RVLIS). EPRI's position for not providing the RVLIS for the passive ALWRs was discussed in Section 2.4.1 of Appendix B to Chapter 1 of the passive Requirements Document as a plant optimization subject. As stated in Section 3.5 of Chapter 3 of this report, the staff requires that all PWR designs have an RVLIS to meet the requirement of 10 CFR 50.34(f)(2)(vii). In Revision 4 of the passive Requirements Document, EPRI added Section 6.3.3.2 of Chapter 4, which requires that instrumentation be provided for monitoring water inventory in the reactor vessel from at least the bottom of the hot legs to the top of the reactor vessel. In addition, Section 2.4.1 of Appendix B to Chapter 1 has been deleted from the list of plant optimization subjects list.

Though not specifically stated in Section 6.3.3.2 of Chapter 4 of the passive Requirements Document that the RVLIS will meet the requirement specified in TMI Action Item II.F.2, "Instrumentation for Detection of Inadequate Core Cooling," of NUREG-0737, EPRI does indicate its intent to meet NUREG-0737 as stated in Table B.1-2 of Appendix B to Chapter 1. TMI Action Item II.F.2 requires that instrumentation provide an unambiguous, easy-to-interpret indication of inadequate core cooling (ICC). Clarification Item (6) of Item II.F.2 states that the indication must cover the full range from normal operation to complete core uncovery, and also provides examples that can be used for this purpose. For example, water-level instrumentation may be chosen to provide advance warning of two-phase level drop to the top of the core and could be supplemented by other indicators such as in-core and core-exit thermocouples, provided the indicated temperatures can be correlated to provide indication of the existence of ICC and to infer the extent of core uncovery. Therefore, whether or not the requirement of Section 6.3.3.2 of Chapter 4 for instrumentation monitoring water inventory in the reactor vessel from the bottom of the hot legs (rather than the bottom of the core) to the top of the reactor vessel meets the requirement of TMI Action Item II.F.2 will depend on other supplemental instrumentation. As detailed instrumentation

design is not available, the staff will review the integrated ICC instrumentation during its review of an application for FDA/DC to ensure that an adequate design is specified for the RVLIS.

The staff's evaluation of EPRI's requirements for the reactor shutdown cooling system is provided in Section 9 of Chapter 3 of this report.

Conclusion

The staff concludes that the requirements in Section 6.3 of Chapter 4, with the exception of the vendor- or utility-specific items noted above, do not conflict with NRC regulations.

7 PWR CORE AND FUEL

7.1 Definition and Scope

The PWR core section will encompass fuel assemblies and fuel rods, reactivity control devices, neutron sources, and core instrumentation. The core and fuel are to generate heat output up to the rated value, throughout planned operating cycles, with sufficient margin and control to accommodate anticipated transients and planned maneuvers, all within defined limits.

7.2 Performance Requirements

7.2.1 Core and Fuel Power Generation

Stable Operation

Section 7.2.1.2 of Chapter 4 of the passive Requirements Document specifies that the core operating characteristics will provide for stable operation for all expected operating conditions and will permit the operator to recognize the imminence of unstable operation during system disturbances and take actions to avoid it. It further requires that thermal-hydraulic stability and xenon stability characteristics of new design be verified by in-plant or other testing. The staff has found that thermal-hydraulic oscillations are not a problem for the current generation of PWR plants. This finding may have to be revalidated if the thermal-hydraulic characteristics of the advanced PWRs are sufficiently different from those of current designs. The staff allows PWRs to have axially unstable, but not radially unstable, xenon oscillations. Therefore, the EPRI requirement of in-plant or other testing to verify thermal-hydraulic and xenon stability characteristics of new designs is consistent with the staff's position.

Margin to Fuel Design and Operating Limits

Section 7.2.1.3.1 of Chapter 4 of Revision 0 to the passive Requirements Document required that the most limiting fuel rod meet the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K, for all postulated loss-of-coolant-accident (LOCA) events. This section appeared to require the LOCA analysis to be restricted to the old evaluation model based on the Appendix K requirements. In its letter dated May 8, 1991, responding to a staff concern transmitted by letter dated March 8, 1991, EPRI deleted the requirement for compliance with Appendix K requirements in Revision 2 of the passive Requirements Document. Therefore, the plant designer can perform a LOCA analysis on the basis of the new emergency core cooling system rule specified in 10 CFR 50.46, which allows the use of realistic best-estimate evaluation methodologies. As stated in Section 4.2.1 of this chapter, the staff concludes that the passive-plant LOCA analyses should be based on best-estimate LOCA analyses.

Load-Following and Maneuvering Capability

Section 7.2.1.4.1 of Chapter 4 requires the core to be designed with the capability for load following and programmed load cycling without adjusting the soluble boron concentration during the maneuver. It further specifies

that changes in moderator temperature would be used for rapid response to a small load change, and low-worth control rods would be used to assist in programmed load-cycling control. Both of these mechanisms have been proven in operating PWRs. In the rationale portion of this section, EPRI states that extrapolation to rodded maneuvering control without the use of soluble boron has been assessed to be feasible and practicable with increased use of proven, low-worth control rod designs. Although, generally not the practice in current PWRs, current NRC requirements would allow such capabilities. However, the staff will evaluate plant-specific analyses during its review of an application for FDA/DC.

Section 7.2.1.4.2 of Chapter 4 requires that the fuel be designed to avoid limitation of maneuvering capability and the rate of power increase for hot startup of the plant, and that restriction of cold startup power due to fuel be eliminated. The staff recognizes the benefits of meeting this objective, but will evaluate the specific fuel designs during its review of an application for FDA/DC.

7.2.2 Fuel Burnup, Reliability, and Lifetime

Section 7.2.2.2 of Chapter 4 requires the basic fuel mechanical design to be capable of assembly-average burnup of at least 60,000 megawatt-days per metric ton uranium (MWD/MTU). This is inconsistent with the 55,000 MWD/MTU requirement of Section 3.6 of Chapter 1. Since the current burnup limit for existing PWR fuel is approximately 55,000 MWD/MTU, to extend this higher fuel burnup limit to at least 60,000 MWD/MTU, the staff requires sufficient data on high fuel burnup to demonstrate fuel integrity in the areas of fission gas release, cladding corrosion due to oxidation and hydriding, and reduction in cladding material strength. EPRI indicates that the intent of this requirement is to ensure that reactor internals and fuel assemblies are designed to accommodate high burnup effects. These requirements are not based on any specific database extending to the specified burnup level. The burnup limit actually placed on ALWR fuel would be that demonstrated and licensable by a specific fuel vendor at the time of use. The staff also discusses this issue in Section 3.6 of Chapter 1 of this report.

7.2.3 Control Rod Assembly Lifetime

Section 7.2.3 of Chapter 4 requires the PWR control rod assemblies to be designed for a minimum operating lifetime of 15 reactor full-power years (RFPYs) with an objective of 20 RFPYs. This requirement is beyond the operating experience data for the current LWRs. To support the desired extended operating life of the control rod assemblies, sufficient performance data will be required to demonstrate structural integrity under irradiation effects, including material hardening, absorber depletion, and swelling. The staff discusses this issue in Section 3.6 of Chapter 1 of this report.

7.3 Equipment Design Requirements

Core and Fuel Nuclear Design

Section 7.3.1.2.3 of Chapter 4 of the passive Requirements Document requires the fuel cycle design to have a non-positive moderator temperature coefficient (MTC) over the entire fuel cycle, at any power level, with the reactor coolant

system at normal operating temperature when the reactor is critical. EPRI states that a non-positive MTC simplifies operational control of the reactor and is considered to be a contributor to safety. However, the staff concludes that too large a negative MTC could be detrimental to certain cooldown transients such as a steamline break. Therefore, a most negative MTC limit, which is determined by plant-specific safety analysis, will need to be specified in plant technical specifications as a limiting condition for operation for the modes from hot standby to power operation.

Materials

Section 7.3.1.4.3 of Chapter 4 requires that control rod materials be proven by successful LWR operating experience. The following materials satisfy that guideline:

Hub and spider assembly	Stainless steel Type 304L or 316L Solution annealed with less than 0.02 percent carbon
Absorber rodlet cladding	Stainless steel tubing Type 304L or 316L Solution annealed with less than 0.02 percent carbon and other special requirements to reduce the susceptibility to irradiation-assisted stress corrosion cracking of cladding intended to undergo some strain. Inconel 625 tubing may also be used. The specific design must take into account strain limits verified through service experience.

Section 7.3.1.4.3 of Chapter 4 also allows absorber materials to vary depending on how the lifetime objective of 20 years is to be achieved with the different control rod designs and duty requirements (e.g., silver-indium-cadmium (Ag-In-Cd) or hybrids with Ag-In-Cd at the high exposure tip and boron carbide in the low-exposure regions can be used). However, it also specifies that clad hafnium will not be used because the potential for swelling of control rods has developed into a safety concern since the phenomenon was first noticed at a PWR in 1988. Experience has shown that clad hafnium is subject to swelling that can lead to control-pin cladding failures and binding of the control rods. Therefore, the requirement to prohibit use of clad hafnium eliminates this concern.

The staff concludes that these requirements are acceptable because the materials specified have performed well in service.

Fuel Rod Bow

Section 7.3.1.10 of Chapter 4 requires the fuel assembly to be designed so that closure of the spacing between fuel rods does not exceed 50 percent of the nominal spacing under operating conditions throughout life. This requirement addresses the concern about the effect of fuel rod bowing. Thermal-hydraulic tests showed that closure of spacing between fuel rods that is more than 50 percent has detrimental effects on critical heat flux and departure from nucleate boiling ratio (DNBR). Therefore, fuel rod bow penalties for a decrease in spacing more than 50 percent have been used for the current fuel design. EPRI states that designing for less than a 50-percent fuel rod gap

closure should preclude unacceptable power peaking or DNBR conditions based on NRC-approved rod bow evaluation models. If EPRI intends to remove the rod bow penalty, passive-plant vendors complying with the passive Requirement Document will have to provide evidence, such as actual fuel data or analyses, supporting the statement that fuel rod closure will be less than 50 percent during fuel life to the staff during its review of an individual application for FDA/DC.

8 PWR CONTROL ROD DRIVE SYSTEM

8.1 Definition and Scope

The functional requirements of the PWR CRD system include (1) position (withdraw and insert) the control rods in the core in response to commands from the rod control system, (2) release the control rods for gravity insertion into the core on power interruption in response to a reactor trip initiated from either manual or automatic reactor protection system controls at the required rate to maintain fuel integrity and, (3) permit the latching and unlatching of the connection between the drive rod and the control rod assemblies.

8.2 Performance Requirements

Section 8.2 of Chapter 4 of the passive Requirements Document requires the CRD system pressure boundary to be designed for a service life of 60 years and establishes the design and test criteria as follows:

operating-basis earthquake: 300 full-load cycles
safe-shutdown earthquake: 1 event
operational scrams: 1500
test scrams: 450
pressure test: 1 per year

These criteria appear to be appropriate for a service life of 60 years. However, since this proposed lifetime exceeds that of the existing CRD mechanisms (CRDMs), it is not known if that goal can be achieved.

In Section 8 of the DSER for Chapter 4 of the passive Requirements Document, the staff stated that although these criteria appear to be appropriate for a service life of 60 years, this proposed lifetime exceeds that of existing CRDMs; hence, it is not known whether that goal can be achieved. EPRI's position is that the Requirements Document establishes the criteria that the CRDMs must be qualified to meet. It is common industry practice with these components to perform lifetime testing to qualify them for the design lifetime. EPRI's requirements will be used as the basis for this testing and plant designers must demonstrate that the CRDMs have been satisfactorily tested to meet EPRI's requirements. The staff concludes that this is acceptable; however, as part of its review of an application for FDA/DC, it will review in detail the results of design and testing programs that will be implemented in accordance with the above criteria to demonstrate that the CRDMs are qualified for a 60-year service life.

Scram Time

Section 8.2.2 of Chapter 4 requires the CRD system to be designed so that, if power to the CRD coils or motors is interrupted, control rods will insert by gravity from the fully withdrawn position to the fully inserted position within a predetermined scram time (including any delays inherent in the associated instrumentation and control system) so that the total time from the

primary sensor response to a change in a monitored parameter for rod insertion satisfies the most restrictive accident analysis. The staff concludes that this requirement is compatible with NRC requirements.

Safety and Reliability

Section 8.2.4 of Chapter 4 specifies that the CRD system must be designed so that no single failure of a component, structure, system function, or service function will prevent the CRD system from performing its safety-related function of preventing rod drop and rod ejection. In addition, the CRDMs are required to be capable of operating without coolant flow (air or water) for a minimum of 30 minutes without damage. This capability provides a reasonable period of time for restoring the system after a loss of the CRDM cooling system.

8.3 Equipment Design Requirements

Section 8.3 of Chapter 4 of the passive Requirements Document specifies equipment design requirements for the PWR CRD system, including structural and mechanical considerations, materials, electrical and instrumentation design, and maintenance and testing. These requirements specify a variety of details intended to achieve high-quality design, reliable operation, and simplified maintenance. The staff concludes that the equipment design requirements for the PWR CRD system are compatible with NRC requirements.

Seals

Section 8.3.1.3 of Chapter 4 requires all CRDM seals to be accessible for repair without having to remove any adjacent CRDMs. Similarly, the CRDM stator coils and all electrical parts are to be replaceable without breaking the primary system pressure boundary and without removing any adjacent CRDMs. The staff concludes that these requirements are likely to reduce personnel exposure and are, therefore, acceptable.

Materials

Section 8.3.2 of Chapter 4 requires that materials used in the construction of the CRDM be selected on the basis of the requirements of Chapter 1 of the passive Requirements Document, taking into consideration the requirements of applicable codes and standards and a history of satisfactory service in LWR CRDM applications.

The staff has also reviewed the materials requirements for PWR CRDMs, as discussed in Section 5 of Chapter 1 of the passive Requirements Document, to ensure that materials in the reactor coolant system and related systems are compatible with their intended service and concludes that the requirements are acceptable.

9 CONCLUSION

The staff concludes that the requirements established in Chapter 4 of the passive Requirements Document for the design of reactor vessels and reactor systems do not conflict with current regulatory guidelines and are acceptable. However, by themselves, they do not provide sufficient information for the staff to determine if the design of the reactor vessels and reactor systems will be adequate. Applicants referencing the passive Requirements Document will be required to demonstrate compliance with the additional guidance in the SRP, or provide justification for alternative means of implementing the associated regulatory requirements.

Therefore, the staff concludes that Chapter 4 specifies requirements that, subject to the resolution of vendor- and utility-specific items, if properly translated into a design in accordance with the NRC regulations in force at the time the design is submitted, should result in a nuclear power plant whose reactor vessel and reactor systems will perform as designed and have all the attributes required by the regulations to ensure that there is no undue risk to the health and safety of the public or to the environment.

APPENDIX A
DEFINITIONS AND ACRONYMS

Appendix A of Chapter 4 of the passive Requirements Document contains definitions of acronyms. The staff has provided a consolidated list of acronyms in Volume 1 of this report.

Chapter 5, "ENGINEERED SAFETY SYSTEMS"

1 INTRODUCTION

This chapter of the SER documents the NRC staff's review of Chapter 5, "Engineered Safety Systems," of the passive Requirements Document through Revision 5. Revision 5 was submitted to the NRC by letter dated October 1, 1993. Additional revisions, which were not included in Revision 5, were submitted by two letters dated June 8, 1993. Chapter 5 was prepared, under the project direction of EPRI and the ALWR Utility Steering Committee, by Bechtel Power Corporation; Commonwealth Edison Company; Duke Power Company; General Electric Company; MPR Associates, Inc.; S. Levy Incorporated; Sargent and Lundy Engineers; Stone and Webster Engineering Corporation; Westinghouse Electric Corporation; and EPRI.

1.1 Review Criteria

Section 1 of Volume 1 of this report describes the review criteria the staff used during its review of Chapter 5 of the passive Requirements Document.

1.2 Scope and Structure of Chapter 5

Chapter 5 of the passive Requirements Document defines the ALWR Utility Steering Committee's requirements for engineered safety systems, which will be used to prevent or mitigate the effects of a spectrum of postulated core damage accidents for passive ALWR designs. These engineered safety systems will use passive means to supply safety injection water, provide core and containment cooling, and perform other functions.

EPRI defines passive means as "natural forces," such as gravity, natural circulation, and stored energy. Pressure accumulators and batteries are examples of stored energy. All active auxiliary systems will be designated non-safety systems.

The ALWR safety design is based on the defense-in-depth concept that consists of accident resistance, core damage prevention, and consequence mitigation. The engineered safety systems will be designed to cope with various transients and accidents for core damage prevention and mitigation. In addition, at least one non-safety auxiliary system will be required to protect the core without reliance on the passive safety systems as a first line of defense for all anticipated transients.

The key topics addressed in the Chapter 5 review include EPRI's ALWR public safety goal, severe-accident prevention and mitigation, severe-accident containment performance criteria, hydrogen generation and control, source-term issues, fire protection, high/low-interface design (intersystem loss-of-coolant-accident (LOCA)), anticipated transients without scram (ATWS), operation of residual heat removal (RHR) system with reduced reactor coolant

system (RCS) inventory, station blackout, core-concrete interaction - capability to cool core debris, high-pressure core melt ejection, and equipment survivability.

1.3 Policy Issues

During its review of Chapter 5 of the passive Requirements Document, the NRC staff did not identify any issues that involve policy questions for the technical areas discussed in this chapter, other than those already identified in the Commission papers listed in Appendix B to Chapter 1 of this report.

1.4 Outstanding Issues

The DSER for Chapter 5 of the passive Requirements Document contained the following outstanding issues:

Open Issues

P.5.0-1 justification for 72-hour design-basis period for control room habitability (2.1.1, 2.2, and 6.5)

P.5.0-2 need for activated charcoal filters (2.1.2)

P.5.0-3 timing of fission product release (2.1.3)

P.5.0-4 evaluation of aerosol fission product removal (2.1.6)

P.5.0-5 secondary building fission product holdup and plateout (2.1.7 and 6.4)

P.5.0-6 chemical form of iodine in containment (2.1.9)

P.5.0-7 guidance on vendor-supplied information (2.4.2)

P.5.0-8 identification of vital equipment (2.5)

P.5.0-9 use of carbon austenitic stainless steel for passive decay heat removal (PDHR) heat exchanger piping material (3.3)

P.5.0-10 diverse reactor protection system input to control rods (3.4)

P.5.0-11 definition of the safe shutdown condition (4.3)

P.5.0-12 PDHR water pool capacity (4.3)

P.5.0-13 gravity drain tank for standby liquid control system (SLCS) (4.5)

P.5.0-14 separate connecting line for the two trains of the automatic depressurization system final stage (5.4)

P.5.0-15 elimination of Type C leakage rate testing (6.2)

P.5.0-16 Type B air-lock leak test interval requirements (6.3)

P.5.0-17 maximum interval for Type C leakage rate testing (6.3)

P.5.0-18 dose consequence criteria for design-basis accidents (6.4)

P.5.0-19 need for safety-grade containment spray system and engineered safety feature atmosphere cleanup (6.4 and 2.1.6)

P.5.0-20 thyroid and beta skin radiation dose limits and credit for long-term use of breathing apparatus (6.5)

P.5.0-21 hydrogen concentration for PWR dry containment (6.6)

P.5.0-22 allowable compressive stress consistent with ASME Code (6.6)

P.5.0-23 safety-grade hydrogen recombiner system (6.6)

P.5.0-24 containment steam bypass capability consistent with Standard Review Plan (7.2)

Confirmatory Issues

None

The final disposition of each of these issues is discussed in detail in the appropriate section of this chapter, as indicated by the parenthetical notation following each issue. All outstanding issues identified in the DSER for Chapter 5 have been resolved.

1.5 Vendor- or Utility-Specific Items

The DSER for Chapter 5 contained the following vendor- or utility-specific items. The final disposition of each of these items is provided in the appropriate section of this chapter, as indicated by the parenthetical notation following each item.

P.5.V-1 decontamination factor for containment system (2.1.8)

P.5.V-2 challenge from inadvertent opening of the depressurization system (2.2)

P.5.V-3 safety-grade provisions for the fire protection system (2.3)

P.5.V-4 LOCA calculations justifying removal of core spray system (4.1)

P.5.V-5 system design to minimize condensation water hammer (4.2)

P.5.V-6 manual SLCS initiation (4.5)

P.5.V-7 in-containment refueling water storage tank boiling suppression (5.3)

P.5.V-8 justification for use of remote manual valve for containment isolation (6.2)

P.5.V-9 evaluation of the ignition system for combustible gas control (6.6)

P.5.V-10 reliability of power supplies for severe-accident equipment (6.7)

P.5.V-11 detailed discussions regarding design-basis-accident events (9)

The following vendor- or utility-specific items remain and will be addressed during the staff's review of an individual application for FDA/DC. The staff's position on each item is provided in the appropriate section of this chapter, as indicated by the parenthetical notation following each item.

P.5.V-1 need for ESF atmosphere cleanup filtration systems (2.1.2)

P.5.V-2 timing of fission product release (2.1.3)

P.5.V-3 evaluation of aerosol fission product removal (2.1.6)

P.5.V-4 secondary building fission product holdup and retention (2.1.7)

P.5.V-5 decontamination factor for containment system (2.1.8)

P.5.V-6 technical bases for emergency operating procedure guidelines (2.2)

P.5.V-7 challenge from inadvertant opening of the depressurization system (DPS) (2.2)

P.5.V-8 balance between severe accident prevention and mitigation (2.4)

P.5.V-9 regulatory treatment of non-safety systems (2.2 and 6.6)

P.5.V-10 design of passive safety injection system (PSIS) (3.2)

P.5.V-11 safe shutdown condition (3.3 and 5.3)

P.5.V-12 compliance with 10 CFR 50.46 (4.1 and 5.4)

P.5.V-13 redundancy and separation requirements for PSIS pool (4.2)

P.5.V-14 potential for condensation water hammer in PSIS (4.2)

P.5.V-15 adequacy of suppression pool driving head (4.2)

P.5.V-16 adequacy of design to provide PDHR for 72 hours without operator action (4.3)

P.5.V-17 adequacy of heat exchanger venting (4.3)

P.5.V-18 adequacy of PDHR system throttling design (4.3)

P.5.V-19 adequacy of DPS (4.4)

P.5.V-20 minimum flow rate for boron mixing (4.5)

P.5.V-21 adequacy of SLCS volume control (4.5)

P.5.V-22 adequacy of SLCS accumulator pressure control (4.5)

P.5.V-23 adequacy of PSIS model adequacy (5.2)

P.5.V-24 CMT overpressue protection (5.2)

P.5.V-25 adequacy of test program to identify phenomena affecting CMT performance (5.2)

P.5.V-26 long-term core cooling capability (5.3)

P.5.V-27 adequacy of PDHR heat transfer (5.3)

P.5.V-28 licensing design-basis analysis of steam generator tube rupture events (5.4)

P.5.V-29 fission product leakage control (6.4)

P.5.V-30 Type C testing for secondary systems that are closed inside the containment (6.2)

P.5.V-31 control room habitability (6.5)

P.5.V-32 acceptability of passive autocatalytic recombiners for combustible gas control (6.6)

P.5.V-33 analyses of postaccident oxygen concentrations (6.6)

P.5.V-34 vulnerability of containment design to local buckling (6.6)

P.5.V-35 core-concrete interaction (6.7)

P.5.V-36 cavity/pedestal-drywell configuration (6.7)

P.5.V-37 severe-accident-equipment requirements (6.7)

P.5.V-38 steam bypass of suppression pool (7.2)

2 POLICY STATEMENTS AND KEY REQUIREMENTS

2.1 Policy Statements

Section 1.2 of Chapter 5 of the passive Requirements Document contains several ALWR program policy statements regarding engineered safety systems for the passive plant. The staff reviewed these policy statements to determine if they conform to current regulatory requirements. When a deviation from current regulatory requirements or guidance was apparent, the staff considered the justifications provided by EPRI.

2.1.1 Seventy-Two-Hour Design-Basis Period for Control Room Habitability

In Section 1.2.1.3 of Chapter 5, EPRI states: "The control room shall be designed such that it can be maintained during the 72 hour design basis period as the primary location from which personnel can safely operate in the case of an assumed accident and other postulated design conditions." Depending on the accident sequences, the "duration of the accident" can be much longer than 72 hours in design-basis accidents. General Design Criterion (GDC) 19 of Appendix A to 10 CFR Part 50 states that "adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions . . . for the duration of the accident." Because EPRI provided no justification for this deviation, the staff concluded in the DSER for Chapter 5 that for the analysis of control room habitability, the duration of the accident not just the EPRI-proposed 72-hour design-basis period should be considered. This was identified as an open issue in the DSER. EPRI has revised its requirements for the design of the control room and now specifies a passive control room pressurization system for 72 hours with only simple operator actions after that period (i.e., replenishing air tanks using safety-grade connections). The resolution of this issue is given in Section 6.5 of this chapter. On the basis of the evaluation in Section 6.5, this open issue is closed.

2.1.2 Filtered Exhaust Ventilation Systems

Section 1.2.3.1 of Chapter 5 states that there will be no safety-grade filtered exhaust systems in the ALWR passive plant ventilation systems. It further states that charcoal filters, which are required solely for the removal of elemental and organic iodine, are complex, have been difficult to operate and test, and will not be necessary because the amount of elemental iodine expected to be released in an accident will be small.

Sections 2.2.3 and 2.4.2 of Chapter 5 require that plant designs demonstrate that dose consequences of licensing design-basis events will not exceed regulatory limits.

In operating light-water-cooled nuclear power plants, atmosphere cleanup systems are included as engineered safety features (ESFs) to mitigate the consequences of postulated accidents by removing from the building or containment atmosphere radioactive material that may be released in the accident. Atmosphere cleanup systems are typically included in the standby gas treatment system and the emergency air cleaning systems for the fuel handling building, control room, and shield building.

A typical ESF atmosphere cleanup system consists of some or all of the following components: demisters, heaters, prefilters, high-efficiency particulate air (HEPA) filters, and charcoal adsorbers. The demister removes entrained water droplets from the inlet stream, thereby protecting prefilters, HEPA filters, and adsorbers from water damage and plugging. Heaters normally follow the demisters in the cleanup train and are designed to heat the incoming air stream to reduce the stream's relative humidity before it reaches the filters and adsorbers. Prefilters and HEPA filters remove particulate matter that may be radioactive. Prefilters remove the larger particles and prevent excessive loading of HEPA filters; to some extent, demisters may also perform this function. The HEPA filters remove the fine discrete particulate matter and pass the air stream to the adsorber. Charcoal adsorbers remove iodine in elemental and organic forms from the air stream. HEPA filters downstream of the adsorption units collect carbon fines and provide redundant protection against particulate release in case the upstream HEPA filter bank fails.

EPRI does not require an ESF atmosphere cleanup system for its passive plant designs. It originally did not submit a radiological accident consequence assessment to demonstrate that without an ESF atmosphere cleanup filtration system, the resulting potential offsite doses would be less than the dose guideline values of 10 CFR Part 100 and the control room personnel potential radiation exposures would be within the requirements of GDC 19. In the DSER for Chapter 5, the staff was particularly concerned about the elimination of charcoal adsorbers from the passive plant design. This was identified as an open issue.

In its letter dated June 22, 1993, EPRI submitted a preliminary radiological accident consequence assessment to demonstrate the feasibility of a passive design without an ESF atmosphere cleanup filtration system. The staff's technical positions on this issue are contained in the draft Commission paper, "Source Term-Related Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water-Reactor Designs," dated February 10, 1994. In this policy paper, the staff evaluates the feasibility of licensing a passive design by allowing credit for natural removal mechanisms inside the containment based on a new physically based source term. The staff concludes, while that such a design appears feasible in concept, plant designers will have to demonstrate that their designs meet all regulatory requirements.

In its letter dated June 8, 1993, EPRI reiterated its position that no ESF atmosphere cleanup filtration system need be provided in the passive plant design, provided passive ALWR vendors submit an acceptable radiological consequence assessment to the staff. The staff finds EPRI's position as stated above to be acceptable, provided the assessment, using the fission-product release timing based on its specific passive LWR design and selected accident sequences and using their best-estimate, in-containment fission-product removal efficiencies, demonstrates that the passive ALWR design meets the dose reference values in 10 CFR Part 100 and the control room operator dose limits specified in GDC 19. The staff will evaluate this radiological consequence assessment during its review of an individual application for FDA/DC. Therefore, this open issue is closed.

2.1.3 Timing of Fission Product Release

Section 1.2.3.3 of Chapter 5 and Table B.2 of Appendix B to Chapter 5 specify core damage event timing. In these sections, EPRI notes that present licensing practice assumes that design-basis-accident fission product releases to the containment atmosphere occur virtually instantaneously. This, EPRI claims results in containment isolation valve closure times that are shorter than necessary and ventilation fan capacities larger than necessary, based on analysis of sequences expected to dominate the likelihood of core damage. The passive Requirements Document takes the position that an accidental release of fission products into the containment is assumed to occur no sooner than about 1 hour after reactor scram. The basis for this assumption is given in EPRI technical reports entitled "Evolutionary Plant Source Term Report (October 1990)" and "Passive ALWR Source Term (February 1991)." These reports stated that the initial fission product release from the reactor coolant system to the containment atmosphere (gap activity) occurs at a time greater than 30 minutes after scram for existing plants. The reports further stated that allowing the enhancements provided by the ALWR design requirements, significant radioactive release (fuel activity) to the containment would occur no sooner than about 60 minutes after reactor scram for a plant designed to meet the ALWR Requirements Document.

For licensing purposes, timing of accident fission product releases into the containment is given by Regulatory Guides (RGs) 1.3 and 1.4, Position C.1.a. These positions indicate that fission product releases should be assumed to be "immediately" available for leakage from the containment. In contrast to the instantaneous releases that have been postulated in RGs 1.3 and 1.4, analyses of severe reactor accident sequences have shown that, despite differences in plant design and accident sequences, such releases can be generally categorized in terms of phenomenological phases associated with the degree of fuel melting and relocation, reactor vessel integrity, and, as applicable, attack on concrete below the reactor cavity by molten core materials. The general phases, or progression, of a severe reactor accident are gap activity release, early in-vessel release, ex-vessel release, and late in-vessel release.

In the DSER for Chapter 5, the staff stated that it believed that definitive relaxation in the assumed timing of large fission product releases into the containment associated with fuel degradation and melting was warranted in terms of these phenomenological phases, but that it could not support the specific value (1 hour after scram) proposed in the passive Requirements Document. When the DSER was being prepared, the staff was completing the update of the TID-14844 source term, including timing of accident fission product releases into the containment. In the DSER, the staff stated that it would complete its evaluation of this issue along with the entire source term-related issues in the passive Requirements Document once the Commission approved the staff's revised TID-14844 source term. This was identified as an open issue.

Draft NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," contains the staff's draft revised source term to reflect a new, physically based term for severe accidents. In draft NUREG-1465, the staff indicated that fission product "gap activity release" was estimated for a large-break loss-of-coolant accident to commence no earlier than 10 to 30 seconds for PWRs and 30 seconds for BWRs. It also indicated that fission

product "early in-vessel release" was estimated to commence no earlier than 0.5 and 1.0 hour for PWRs and BWRs, respectively. In contrast to the instantaneous releases that were postulated in RGs 1.3 and 1.4, analyses of severe-accident sequences have shown that fission product releases can be generally categorized in terms of phenomenological phases associated with fuel failure, melting, and relocation despite differences in reactor design and accident sequence.

The selected release times and durations shown in draft NUREG-1465 were chosen primarily on the basis of simplicity and consistency. The staff will use the fission product release times and durations given in draft or final NUREG-1465, unless the ALWR vendor provides acceptable alternative fission product release times and durations based on its reactor design and applicable accident sequences chosen. Preliminary information provided by EPRI indicates, that for the passive plant designs, release of fission products from the fuel to the containment atmosphere is expected to begin about one hour after the scram of the reactor. An accurate determination of the release times and durations depends not only on the reactor design but also on the selected accident sequences. The staff will evaluate accident sequences, timing, and duration submitted by passive plant vendors for the purpose of determining fission product releases for a given reactor design during its review of an individual application for FDA/DC. Therefore, this open issue is closed.

2.1.4 Containment Leakage Rates

Section 1.2.3.4 of Chapter 5 states that the containment leakage varies as a function of containment pressure. In addition, this section states that the containment leak rate assumed for the licensing design-basis radiation dose calculations must be 0.5 weight percent per day at design pressure in order to provide more operating flexibility for containment leak rate testing and associated maintenance, while still meeting 10 CFR Part 100 radiation dose criteria for licensing design-basis events. Section 1.2.3.4 allows smaller values for PWRs (no lower than 0.1 percent per day), provided the plant designer can identify offsetting benefits.

Current licensing practice is that the containment design leak rate is not a fixed value, but is determined as the value that, in combination with other plant and site parameters, will result in calculated radiation doses not exceeding the dose reference values specified in 10 CFR Part 100 as a result of the accident postulated and evaluated using RGs 1.3 and 1.4. Typical containment design leak rates have been 0.1 to 0.2 weight percent per day for single-containment PWRs and about 0.5 weight percent per day or more for BWRs with typical standby gas treatment and main steam isolation valve (MSIV) leakage control systems. RG 1.4 specifies that for PWRs the containment leak rate is assumed to remain constant for 24 hours, before decreasing to half the design leak rate. For BWRs, RG 1.3 specifies that the containment leak rate remains constant for the duration of the accident.

The staff has reviewed EPRI's position on containment leakage rate varying as a function of containment pressure. The staff recognizes that the current guidance of constant or near-constant containment leak rates in RGs 1.3 and 1.4 provides substantial margins when compared to varying the leakage rate as a function of pressure. However, since the RGs were published, more has

become known about the effects of containment pressure on the leakage rate. The staff has determined that pressure curves can be conservatively developed that provide adequate safety margins without resorting to the restrictive guidance of RGs 1.3 and 1.4. Therefore, EPRI's position on containment leakage varying as a function of containment pressure is acceptable, provided leak tests are performed periodically at pressures that bound the containment design pressure profiles and provided there is an adequate safety margin.

The staff considers that the 0.1 to 0.5 weight percent per day containment leakage rate will have to be evaluated for offsite and control room radiological consequences to meet 10 CFR Part 100 requirements based on the revised source term. Features of the containment structure and its associated penetration designs could minimize actual leakage to much less than the allowable leakage limit. Therefore, the staff concludes that the proposed allowable containment leakage of 0.1 to 0.5 weight percent per day is acceptable, provided the calculated radiological consequences for design-basis accidents meet the radiation dose reference values in 10 CFR Part 100 and the control room operator radiation dose limits in GDC 19 of Appendix A to 10 CFR Part 50.

2.1.5 BWR MSIV Leakage Control System

Section 1.2.3.5 of Chapter 5 states that the MSIV leakage control system will not be provided in the passive BWR. The staff evaluates this issue in Section 2.3.1 of Appendix B to Chapter 1 of this report.

2.1.6 Primary Containment Aerosol Removal

Sections 1.2.3.6 and 6.4.1 of Chapter 5 states that the passive systems that are specified for containment decay heat removal are effective means for removing aerosol fission products and that this capability will be evaluated and utilized in defining the suspended fission product concentration that is available to leak from the primary containment.

The principal means of removing the airborne fission products from the containment atmosphere in the LWR designs traditionally included use of active containment atmosphere cleanup systems such as containment spray, ESF filtration, and pressure-suppression pool scrubbing. In the TID-14844 source term, the staff assumed 50 percent of radioactive iodine (91 percent in the form of elemental iodine) released into the containment to be immediately removed by plateout (diffusion mechanism). In the passive ALWRs, an active containment atmosphere cleanup system will not be used. Reliance is placed solely on such natural aerosol removal processes in the containment as holdup (for decay), sedimentation (for settling), diffusion (for plateout), and leakage (for depletion).

One mechanistic treatment of aerosol behavior in the reactor containment is found in CONTAIN, a computer code developed at Sandia National Laboratories under NRC sponsorship for the analysis of containment aerosol behavior under severe-accident conditions. Another computer code, NAUA, which is very similar to CONTAIN, was developed at Kernforschungszentrum, Karlsrhue, Federal Republic of Germany, and was used for aerosol treatment in the NRC Source Term Code Package. There are a number of other aerosol behavior computer codes, but these two codes are the most widely used and accepted throughout the

international nuclear safety community. Either code is acceptable to the staff for the evaluation of aerosol fission product behavior in the primary containment following a severe reactor accident. However, most of these computer codes use lumped parameter methods, which assume the aerosols to be well mixed in the containment. When the condition of mixing cannot be established, the calculated results using the above computer codes may not be valid.

The staff will consider two natural processes that remove aerosols from the containment atmosphere over the entire period of an accident (30 days): (1) sedimentation mechanism of gravitational settling, including aerosol agglomeration, and (2) diffusion mechanisms of diffusiophoresis and thermophoresis. Both diffusiophoresis and thermophoresis cause the deposition of aerosol particles on all surfaces (e.g., walls and ceiling, as well as the floor) regardless of their orientation. Diffusiophoresis is the process by which water vapor in the atmosphere collects aerosol particles as it migrates (diffuses) toward a relatively cold surface on which condensation is taking place. Thermophoresis also causes aerosol particles to move toward and deposit on colder surfaces, but not as a result of mass motion. Rather, the decreasing average velocity of the surrounding gas molecules tends to drive the particle down the temperature gradient until it traverses the interface layer and comes into contact with the surface where it sticks.

The staff made a preliminary survey of aerosol removal rates from the containment atmosphere on the basis of a review of a selected number of severe-accident sequences with the assumption that the aerosol concentration within the containment decreases with time at a simple exponential rate. In the DSER for Chapter 5, the staff stated that it would complete this evaluation of aerosol fission product removal in the ALWR primary containment once the Commission approved the staff's revised TID-14844 source term. This was identified as an open issue. The staff also identified the elimination of a containment spray system from the passive design as an open issue, because containment sprays are used in current PWR designs to provide postaccident containment cooling as well as to remove radioactive aerosols that have been released. Sprays are highly effective in reducing the airborne concentration of elemental and particulate iodines as well as other particulates, such as cesium iodide. The reduction in airborne radioactivity within the containment by a spray system as a function of time is expressed as an exponential reduction process in which the spray removal coefficient is taken to be constant over a large part of the regime. Typical PWR containment spray systems are capable of rapidly reducing the concentration of airborne activity (by about two orders of magnitude within periods of about 20 to 30 minutes, where both spray trains are operable) within the containment.

For this reason, the staff concluded in the DSER for Chapter 5 that a primary containment spray system was required unless EPRI could demonstrate that the passive ALWR design (with a bounding reference site parameter) without a containment spray system could meet the dose reference values in 10 CFR Part 100 and the control room operator dose limits in GDC 19.

In a paper on the source term for passive plants supplemented with a technical report titled "Passive ALWR Containment Natural Aerosol Removal" (April 1993), EPRI has provided radioactive aerosol removal rates based on sedimentation and diffusion of aerosols in the containment atmosphere using the EPRI version of

the NAUA code (NAUAHYGROS). The staff agrees with EPRI on the physical processes associated with natural aerosol removal that could be taken into account in establishing the airborne fission product concentrations in the containment atmosphere. Therefore, it accepts the aerosol removal mechanisms proposed by EPRI in the passive Requirements Documents and concludes that the calculational methodology employed appears reasonable. However, the actual containment aerosol removal rates (removal coefficients) are plant design specific and will vary, depending on, but not limited to, the containment geometry, containment size, surface area, steam quality, and containment cooling mechanisms. The staff concludes that removal of fission products from the containment atmosphere by natural removal processes such as gravitational settling and thermophoresis, without reliance on an active spray system, is acceptable provided a design-specific analysis demonstrates that natural removal processes will result in offsite doses that meet the values of 10 CFR Part 100 and that doses to the control room operators meet the requirements of GDC 19. Therefore, the amounts of radioactive aerosol deposition in the primary containment will be determined on a case-by-case basis. The specific aerosol removal rates used by the ALWR vendors in conjunction with representative meteorological conditions to demonstrate compliance with the requirements of GDC 19 and 10 CFR Part 100 will be evaluated by the staff during its review of an individual application for FDA/DC. Therefore, the open issues associated with aerosol fission product removal and the need for a safety-grade containment spray system and ESF atmosphere cleanup system are closed.

The staff's technical positions on these two related issues are contained in the draft Commission paper, "Source Term-Related Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water-Reactor Designs," dated February 10, 1994. This paper is contained in Annex C of Appendix B to Chapter 1 of this report.

2.1.7 Secondary Building Fission Product Holdup and Removal

In Section 1.2.3.7 of Chapter 5, EPRI originally stated that the secondary building provided a means of mitigating fission product leakage from the primary containment and that the credit for holdup and removal capability (e.g., equivalent to that achieved from a building turnover rate of 200 percent per day) would be taken for radiation dose reduction. In the DSER for Chapter 5, the staff recognized the holdup phenomenon in the secondary building, but was concerned about the uncertainty in quantifying this phenomenon, validation of calculation models, possible non-uniform distribution of fission product in the building, and adequacy of the leakage control program for the building turnover (leakage) rate during operation. Therefore, the staff determined that using this fission product holdup and plateout in the secondary building of passive plants was, without further justification, not acceptable and identified this as an open issue.

In Revision 5, EPRI revised Section 1.2.3.7 of Chapter 5 to require that the ALWR (BWR) secondary building provide a means of mitigating fission product leakage from the primary containment. EPRI considers that the structure or structures specified in Section 6.4 of Chapter 5 can be classified as a secondary building for the purpose of fission product control and, therefore, for the mitigation of radiological consequences. The passive Requirements Document takes no licensing credit for holdup and retention of fission products in the PWR secondary building but does allow a credit for PRA

and PAG dose evaluations. However, Section 6.4.1 of Chapter 5 the passive Requirements Document states that the appropriate portions of BWR auxiliary and fuel handling buildings that are qualified for the fission product leakage control (FPLC) envelope will be credited for fission product holdup and retention.

The staff accepts EPRI's requirement that the BWR secondary containment provide a means of mitigating fission product leakage from the BWR primary containment. During its review of an individual application for FDA/DC or for a COL, the staff will allow appropriate fission product holdup for decay without requiring that a negative pressure be maintained in the BWR secondary containment if the ALWR (BWR) vendor or the COL applicant (1) demonstrates by testing that the secondary containment leakage and mixing performance are acceptable to the staff and (2) incorporates appropriate periodic leakage testing in the plant technical specifications to validate the claimed leakage rate. Therefore, this open issue is closed. Further details of the staff's position are contained in the draft Commission paper dated February 10, 1994, which is provided in Annex C of Appendix B to Chapter 1 of this report.

The staff also agrees with EPRI that no licensing credit is appropriate for fission product holdup and retention in the PWR secondary building. The staff does not believe it is feasible to qualify the PWR auxiliary and fuel handling buildings for FPLC to licensing calculations. However, if the ALWR (PWR) vendor or the COL applicant can meet the criteria stated above for the BWR secondary containment, the staff will allow appropriate fission product holdup credit for the PWR auxiliary and fuel handling buildings.

These staff technical positions on this issue are contained in the draft Commission paper dated February 10, 1994, which is provided in Annex C of Appendix B to Chapter 1 of this report.

2.1.8 BWR Suppression Pool Radionuclide Removal

In Section 1.2.3.2 of Chapter 5, EPRI indicates that fission product scrubbing capability will be provided to the extent required for severe accidents and this capability will be provided by existing mitigation systems for licensing design-basis events. Credit is taken for the capability of the suppression pool to scrub the containment atmosphere for BWRs. Although the suppression pool is an effective scrubber for both design-basis accidents or severe accidents, the passive Requirements Document does not include any unique requirements for the suppression pool to mitigate severe accidents beyond the normal DBA requirements. The BWR suppression pools are designed primarily as a containment pressure and temperature suppression mechanism for reactor pressure vessel blowdown. However, they can also serve as a medium for scrubbing radioactive fission products except noble gases and iodine in organic forms. The scrubbing (attenuation) of fission products in suppression pools is usually expressed as a "decontamination factor" (DF), which is defined as the ratio of the radioactive material injected into the pool divided by the airborne radioactive material that leaves the surface of the pool water.

RG 1.3, issued in 1974, does not allow credit for fission product scrubbing by BWR suppression pools. However, the Reactor Safety Study (WASH-1400), issued in 1975, assumed a DF of 100 for subcooled suppression pools and 1.0 for

steam-saturated pools. Since 1975, the staff and EPRI have developed detailed models for the analysis of radioactive aerosol scrubbing by the suppression pool. Accordingly, in 1988, the staff issued revised SRP Section 6.5.5, "Pressure Suppression Pool as a Fission Product Cleanup System," stating that ignoring the large volume of research data supporting pool credit would be an undue degree of conservatism. The revised SRP section allowed a maximum DF of 10 for particulate and elemental iodines for a Mark II and a Mark III containment and a DF of 5 for a Mark I containment. These DF values were based on the TID-14844 source term, which stipulates an instantaneous release of fission products from the reactor coolant system to the containment atmosphere. In the revised SRP section, this was further subjected to the fraction of the drywell atmosphere bypassing the suppression pool by leaking through drywell penetrations to obtain the overall DF. The staff has revised TID-14844 by using a physically based source term, as described in draft NUREG-1465.

Therefore, the staff will accept an appropriate suppression pool DF only if the ALWR vendor or COL applicant performs a plant-specific analysis for suppression pool bypass leakage analysis using the fission product release timing and its release durations given in NUREG-1465 or those based on the plant-specific design and applicable accident sequences used for the purpose of determining fission product releases.

Further details of the staff's position concerning suppression pool scrubbing are contained in the draft Commission paper dated February 10, 1994, which is provided in Annex C of Appendix B to Chapter 1 of this report.

2.1.9 Fission Product Chemical Form

The EPRI ALWR passive ALWR source term white paper discusses the chemical forms of radioactive iodines following a reactor accident. It states that 97 percent of the core iodine inventory would exist as nonvolatile particulate species, 2.85 percent as elemental iodine, and the remaining 0.15 percent as organic species.

Using the quantitative results of seven severe reactor accident sequences for LWR nuclear power plants, an NRC contractor analyzed the chemical forms of iodine entering the primary containment following a severe reactor accident. These sequences represented a wide range of risk-significant conditions. The contractor's findings were published in NUREG/CR-5732, "Iodine Chemical Forms in LWR Severe Accidents," dated April 1992. Studies have been completed recently on the chemical form of iodine and its subsequent behavior after it enters the containment from the reactor coolant system (RCS) following a reactor accident. NUREG/CR-5732 and NUREG/CR-5950, "Iodine Evolution and pH Control," document the results of these studies.

In most of the contractor's calculations for the seven sequences, iodine entering the containment from the RCS was almost entirely in the form of cesium iodide (CsI) with very small contributions of elemental iodines (I and hydrogen iodide (HI)). The contractor found that the largest fraction of iodine in forms other than CsI was a total of 3.2 percent as I plus HI. On this basis, the staff preliminarily concluded that iodines entering the containment from the RCS following a reactor severe accident would be composed of at least 95 percent CsI, with no more than 5 percent I plus HI. Because it

had not completed its revision of the TID-14844 source term, the staff identified the chemical form of iodine entering the primary containment following a severe reactor accident as an open issue.

Subsequently, the staff completed its update of TID-14844 and in draft NUREG-1465 concluded that iodine entering the containment from the reactor core is composed of at least 95 percent CsI with no more than 5 percent of I and HI. Once within the containment, highly soluble CsI will readily dissolve in water pools forming iodide (I^-) in solution and deposit onto the interior surfaces. The staff also stated in NUREG-1465 that the radiation-induced conversion of iodide (I^-) in water into elemental iodine (I_2) is strongly dependent on pH. Without pH control, the staff indicated that large fractions of iodine dissolved in water pools in ionic form could be converted to elemental iodine and will be released into the containment atmosphere if the pH is significantly less than 7. On the other hand, if the pH is maintained at 7 or above, very little (less than 1 percent) of the dissolved iodine will be converted to elemental iodine. The passive Requirements Document and all ALWR vendors require that the pH of the water in the containment be maintained at or above 7 (alkaline state) for the entire accident duration to minimize the formation of elemental iodine in the containment water to reduce subsequent release of iodine into the containment atmosphere. The staff agrees with this requirement.

In draft NUREG-1465, the staff did not evaluate the formation of organic iodide in the containment following an accident. It realizes, however, that organic iodide can be produced by the reaction of fission product iodine with organic materials present in the containment. The staff estimates that no more than 5 percent of the airborne elemental iodine will be converted into organic species. This amount of organic iodine would thus correspond to about 0.25 percent of the core iodine inventory (i.e., 5 percent of 5 percent is 0.25 percent). Final NUREG-1465 will address this issue.

EPRI proposed the use of 0.15 percent organic iodine based on 3 percent (i.e., 5 percent of 3 percent is 0.15 percent) of elemental iodine in the EPRI source term. The staff agrees with EPRI on the use of the 5-percent conversion of elemental iodine to organic iodine. However, as explained above, the staff concludes that 5 percent elemental iodine is the appropriate value to which the conversion should be applied. In addition, the staff found in its radiological consequence assessment that iodine in organic form causes the most significant offsite dose because no removal mechanisms are available for this form of iodine other than decay and passive deposition within the containment.

The staff concludes that the chemical forms of fission products following a reactor accident should be at least 95 percent cesium iodide as stated in draft NUREG-1465 with 4.75 percent elemental iodine and hydrogen iodide, and 0.25 percent organic iodine (instead of 5 percent elemental iodine and hydrogen iodide as stated in draft NUREG-1465). EPRI has agreed to the use of these values by the staff until final NUREG-1465 is published and has provided extensive comments to the NRC recommending revisions this to NUREG-report. For BWRs, EPRI's position is that the amount of elemental iodine present and the resulting levels of organic iodine, are substantially lower than those for PWRs and should be considered during the review of the passive BWR. The

staff's technical position on iodine chemical forms is contained in the draft Commission paper dated February 10, 1994, which is provided in Annex C of Appendix B to Chapter 1 of this report. Therefore, this open issue is closed.

2.2 General Design Requirements

Section 2 of Chapter 5 of the passive Requirements Document specifies the top-tier requirements for core damage prevention and mitigation systems and classifies BWR and PWR passive safety systems as core damage prevention systems or as mitigation systems. The core damage prevention functions include the core coolant inventory function, the decay heat removal function, the diverse reactivity control function, and the RCS pressure control function. Section 2.1.3 states that core damage prevention relies not only on the direct functions provided in the engineered safety systems, but also on the support functions provided by non-safety auxiliary systems and on safe operation of normal plant operating systems. The mitigation functions include the containment integrity function and the fission product control function.

Section 2.2.1 of Chapter 5 specifies that the engineered safety systems will perform their assigned functions for all licensing design-basis events. This requirement is consistent with the NRC requirement that only credit for safety-grade systems can be taken in the safety analyses of licensing design-basis transients and accidents.

Section 2.2.2 of Chapter 5 specifies that the designer must demonstrate, using probabilistic risk assessment (PRA), that the plant design meets the ALWR safety goal requirements, that is:

- Frequency of core damage must be less than 1E-5 event per reactor-year.
- Whole-body radiation dose at an assumed 0.5-mile site exclusion area boundary must be less than 25 rem for events whose cumulative frequency exceeds 1E-6 per reactor-year.

In SECY-89-102 and SECY-90-016, the staff recommended that for implementation of the Safety Goal Policy for future standardized plants, the overall mean frequency of a large release of radioactive materials to the environment from a reactor accident be less than 1 in 1,000,000 per year of reactor operation where a large release is defined as one that has a potential for causing an offsite early fatality. The staff has no objection to EPRI's more stringent safety goal. However, it will not use either goal as a staff acceptance criterion.

Section 2.2.4 of Chapter 5 requires the engineered safety systems design to be simplified to make the operator's procedures less complex and interdependent and to reduce the need for protective interlocks. In Section 2.3.4 of Chapter 1 of this report, the staff indicates that the passive Requirements Document design objectives related to emergency operating procedures are acceptable; however, the technical bases for the severe accident management and emergency operating procedure guidelines will be reviewed during its review of an individual application for FDA/DC.

Sections 2.2.9 and 1.2.1.2 of Chapter 5 specify that the probability of actuation of the depressurization system (DPS), which results in release of

reactor coolant directly to the containment atmosphere, will be less than 10 percent over a 60-year plant life, or about 2E-3 event per year. This system reliability requirement is aimed at minimizing challenges to the safety systems and to limit the effect of inadvertent safety system actuation on overall plant availability. In response to a staff question about the basis, EPRI stated that the frequency of 2E-3 event per year was selected because this is about equal to the frequency of small-break LOCA events that would require depressurization actuation for plant protection. Accordingly, EPRI concluded that requiring a substantially lower frequency of inadvertent depressurization would not have a significant effect on the number of actual depressurization actuations occurring over plant life. The staff notes that having a frequency of inadvertent DPS actuations similar to that of small-break LOCAs could significantly add to the probability of the release of reactor coolant to the containment.

The staff recognizes the need for a highly reliable depressurization system to ensure plant depressurization to ensure operation of the passive safety injection system when needed. This need has to be balanced by the possibility of an inadvertent opening of the DPS and its subsequent challenges to the plant safety systems and the containment. The staff will assess this balanced requirement during its evaluation of the probabilistic risk assessment included in an individual application for FDA/DC.

Sections 2.2.10 and 2.2.11 of Chapter 5 originally specified that passive engineered safety systems equipment would be designed and qualified for operation as required in the environment expected during the 72 hours following an assumed accident and coincident loss of ac power and would not require support functions from non-safety systems to provide safety functions for 72 hours. In the DSER for Chapter 5, the staff stated that this ALWR policy and objective of using a 72-hour timeframe as a design basis for the passive system capability was an open issue, as discussed in Chapter 1 of this report.

Sections 2.2.10 and 2.2.11 were revised in Revision 3 of the passive Requirements Document. Section 2.2.10 now specifies that for licensing design basis events, the passive safety systems must have the capability to perform their safety functions and maintain the plant in a safe state with the initial inventory of coolants, air supply, battery capacity, and heat sinks of the safety-grade systems, and without credit for non-safety systems, for at least 72 hours. It further requires that the passive safety systems continue to be available for an extended period to perform their safety function by use of onsite facilities and by connecting readily available offsite equipment and resources to replenish these inventories and provide support as may be necessary. The onsite facilities and the connections to utilize offsite support must be designed to be readily accessible and to withstand safe shutdown earthquake conditions. Offsite equipment and resources required to maintain the plant in a safe state must be shown to be readily available from normal commercial sources or from a prearranged source not subject to the same external event that may occur at the site, and to be placed in readiness at the plant within the initial 72-hour period, as needed.

Section 2.2.11 now requires that plant designers define the recovery process to achieve cold shutdown, and that the passive engineered safety systems equipment be designed and environmentally qualified for the extended period

until the recovery is ensured. Also, passive safety systems must have the capability to be placed back into operation, should their operation be terminated before the end of the recovery period.

The staff finds these requirements to be acceptable, provided the reliability/availability (R/A) of these onsite and offsite supporting equipment and resources are such to ensure safety functions of the passive systems for the time needed. As discussed in Section 2.2 of Chapter 3 of this report, the 72-hour design basis for passive safety system capability will be evaluated as an integral part of an acceptable process for evaluating the regulatory treatment of non-safety systems (RTNSS). This RTNSS process, described in EPRI's May 26, 1993, letter, and implemented in Section 1.6 of Appendix A to Chapter 1 of passive Requirement Document, provides structured steps to be used by plant designers for determining the risk significance and R/A missions of non-safety systems. Appropriate regulatory oversight of these systems, such as reliability assurance program, administrative control, and technical specifications, will be developed on the basis of their risk significance and R/A missions. Therefore, the staff will verify the appropriate R/A and regulatory oversight of the onsite and offsite supports needed to replenish the passive safety system inventories for extended operation during its review of individual applications for FDA/DC.

2.3 Core Damage Prevention Requirements

Section 2.3.1 of Chapter 5 of the passive Requirements Document requires appropriate redundancy of safety-related components and features. EPRI requires that the degree of redundancy for safety-related functions, at a minimum, be sufficient to ensure that safety functions can be accomplished, assuming the most limiting single failure has occurred in addition to the failure that constitutes an accident.

Section 2.3.2 of Chapter 5 requires independence and separation of redundant components "except for areas in which it is physically impractical or less safe, such as in the control room and at the reactor vessel." Cross-connections are not generally allowed. Requirements for electrical separation are given in Chapters 10 and 11. Requirements for mechanical separation include the following:

- Fire protection as specified in 10 CFR 50.48. A 3-hour fire barrier will be used to separate redundant safe shutdown trains, except inside the containment and in the control room. In its letter dated July 22, 1991, EPRI clarified that the provisions in Chapters 6 and 9 of the passive Requirements Document help ensure that these barriers will not delay access of operators or maintenance personnel between components of redundant safety systems. The staff concludes that this is acceptable from an emergency access perspective.

- Physical protection from external and internal flooding.

- Physical protection from pipe breaks inside and outside the containment.

- Spatial separation of redundant components to reduce the probability of common-cause failure.

Section 2.2.3 of Chapter 5 requires that the capability to depressurize and cool down the primary system using safety-grade equipment be provided, assuming an initiating event and the most limiting single failure. This is consistent with the resolution of Policy Issue II.I, "High-Pressure Core Melt Ejection," of SECY-93-087, which requires a depressurization system, and is acceptable.

Section 2.3.4 of Chapter 5 requires that operators (1) be provided with sufficient information, which must be determined by direct measurement of the parameters of interest, to assess core cooling conditions to perform the passive safety injection (PSI) and passive decay heat removal (PDHR) functions and (2) be alerted by unambiguous alarms to the need for manual action to provide the PSI or PDHR function in abnormal situations. The staff concludes that this information (provided via alarms and level instrumentation) should be available at all times, including low-power and shutdown operations, if the non-safety-related shutdown cooling system should become unavailable. This conclusion is consistent with the EPRI requirements in Section 2.12.2.1 of Appendix A to Chapter 1 of the passive Requirements Document. In response to a staff request for additional information, EPRI stated that there were no scenarios that indicated manual actions would be needed to provide the PSI and PDHR functions, but that it was important for operators to have sufficient information to provide essential defense against an unexpected progression of events. Regarding a concern about the possible failure of automatic initiation of the PSI and PDHR functions, Section 8.2.5.4 of Chapter 10 of the passive Requirements Document requires that the man-machine interface systems be designed so that no single failures will prevent or cause the initiation of safety systems and that one additional failure will not prevent safety system actuation. The staff concludes that these requirements satisfy the GDC requirement that a safety system be able to accomplish its safety function assuming a single failure and are, therefore, acceptable.

Section 2.3.5 of Chapter 5 requires that the safety-related core cooling systems provide adequate inventory for core cooling to meet the fuel cladding temperature limits in 10 CFR 50.46 for licensing design-basis events, with a margin as specified in Section 2.2.5 of Chapter 4 of the passive Requirements Document.

Section 2.3.6 of Chapter 5 requires that, on a best-estimate basis, no fuel damage be predicted to occur for a postulated near-instantaneous pipe break with an area equivalent to up to 6 inches in diameter in the reactor coolant boundary. This requirement is intended to provide protection for the greater part of the range of probable pipe breaks.

2.4 Mitigation Requirements

2.4.1 Conformance to Licensing Design-Basis Criteria

Section 2.4.2.6 of Chapter 5 states that the plant design will have the capability to ensure that necessary accident prevention and mitigation functions can be performed during and after events in which hydrogen is produced.

Section 2.4.2.10 of Chapter 5 states that the design will ensure that the pH in containment water pools into which fission product iodine could be transported after an accident will be maintained at a high enough level that formation of elemental iodine from radiolysis in water pools does not preclude meeting ALWR off site radiation dose limits. The staff considers this requirement acceptable because the pH in the containment water pools will be maintained at a high enough level to prevent the formation of elemental iodine.

2.4.2 Conformance to Safety Margin Basis Criteria (Severe Accident Management Program)

Section 2.4.3 of Chapter 5 specifies the ALWR requirements regarding the severe-accident mitigation program. It requires best-estimate analyses to be performed to ensure an adequate margin in the design with regard to severe-accident mitigation and to minimize the reliance placed on active equipment inside the containment. Section 2.4.3.4 requires the plant designer to (1) identify existing systems and equipment that may be useful as part of the severe-accident management program, including both safety and non-safety equipment on site as well as off site; (2) identify actions that may be taken to prevent and mitigate the effects of severe accidents, including preventing core damage, recovering from core damage without vessel failure, maintaining containment integrity, and minimizing offsite radiation release; and (3) review the capability of equipment, including duration of possible use and possible alternative pathway, operator interface information and control, and equipment survivability.

As part of an FDA/DC review for the ALWR designs, the staff will make a determination on the designer's overall treatment of severe-accident issues to ensure that these issues or concerns have been appropriately considered during the design process. Complete consideration of an ALWR design's severe-accident capabilities would include discussion of both design elements that reduce the likelihood of core damage and features that provide accident mitigation if a degraded core event occurs. In the areas of accident prevention, design features that reduce vulnerability to station blackout or provide alternative means of maintaining core decay heat removal, as are provided by the ac independent PDHR systems, would reduce potential vulnerabilities to core damage scenarios. Other design enhancements may reduce potentially significant severe-accident initiators such as fine motion rod drives to reduce the likelihood of ATWS sequences. In the area of mitigation, containment designs could be provided that are expected to retain their integrity for extended periods following a postulated core damage event. Such issues as automatic depressurization system functional reliability to reduce potential containment challenges resulting from direct containment heating and the potential for containment bypass resulting from steam generator tube creep rupture also must be considered.

In its evaluation of the passive ALWR designs, the staff will document the treatment of severe-accident issues. However, in the DSER for Chapter 5, it stated that the passive Requirements Document did not provide any guidance on the scope or content of documentation that the vendors should provide to demonstrate that their design provides an appropriate balance between severe-accident prevention and mitigation. The staff stated that the passive Requirements Document could be improved by providing such additional guidance

to ensure that the design application submittal contains sufficient information to allow the staff to document treatment of these issues. This was identified as an open issue. In its letter dated September 29, 1992, EPRI referred the staff to Section 2.3.3.9 of Chapter 1. The staff concludes that Section 2.3.3.9 provides an adequate overall requirement; however, it lacks any detailed guidance. EPRI has stated that "providing detailed requirements on submittal of detailed documentation to the NRC for beyond licensing basis events is outside the scope of the passive Requirements Document." The staff will review passive ALWR designs for an appropriate balance between severe-accident prevention and mitigation during the review of an individual application for FDA/DC. Therefore, this open issue is closed.

2.5 Sabotage Protection Considerations

Chapter 5 of the passive Requirements Document includes few design requirements that explicitly address protection against sabotage. Requirements for separation of redundant components, physical security, and assessment of the vulnerabilities of the designs to sabotage are included in Chapters 6 and 9 of the passive Requirements Document. In its Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants, dated August 8, 1985, the Commission stated, "The issues of both insider and outsider sabotage threats will be carefully analyzed and, to the extent practicable, will be emphasized in the design and in the operating procedures developed for new plants." Accordingly, the staff has reviewed Chapter 5 to determine how the passive engineered safety system design provisions could affect the inherent sabotage resistance of these designs. In addition, the staff considered whether passive engineered safety system design requirements might introduce any novel vital components that would need protection. It also addressed the possible effect on safety of safeguards requirements contained in Chapter 5.

Section 2 of Chapter 5 includes the general top-level requirements that passive engineered safety system designs must be able to maintain core cooling for 72 hours after initiation of a design-basis accident without any offsite or onsite ac power sources, without support from non-safety-related support systems, and without requiring operator actions. According to EPRI, effective implementation of these top-level requirements would enhance the inherent sabotage resistance of the ALWR.

PDHR systems for the passive PWR will include PDHR system heat exchangers submerged in an in-containment refueling water storage tank (IRWST), which will be open to the containment atmosphere, and a passive containment cooling system (PCCS). The PCCS will include a steel containment shell within a concrete shield building and an air flow intake and chimney that will provide for air cooling of the steel shell's exterior. Both the PDHRS and the PCCS must operate for transfer of decay heat to the ultimate heat sink (outside air) if only the passive systems are available. The PDHRS for the passive BWR will be directly cooled by outside air and also will provide the long-term containment cooling function. Aside from the PDHR pools in which the BWR PDHR condenser tubes will be submerged and the passive containment cooling water tank for the initial phase of the PWR PCCS, no other water supplies outside the containment, such as refueling water storage tanks, condensate water storage tanks, or service water, will be required for support of the passive engineered safety systems. Actuation valves for these systems fail open on loss of control or motive power. The staff considers that these features

increase the inherent resistance of these passive designs to the design-basis threat of radiological sabotage postulated in 10 CFR 73.1(a)(1). In addition, conventional active decay heat removal systems powered by Class 1E power supplies will provide redundant and diverse capability for preventing core damage.

Section 4.3.3 of Chapter 5 specifies that the BWR PDHR pool and condensing heat exchanger will be located outside the primary containment but inside a "structure adequate to provide the physical protection required for a safety-related system." Since the pool will be vented to the atmosphere, a breach of a condenser tube (at full reactor pressure when the PDHR system is not isolated from the RCS) could result in a LOCA outside the containment. The PWR PCCS water storage tank and exterior of the steel containment shell also must be open to the atmosphere for accomplishing the PCCS function. Section 8.3 of Chapter 5 requires access ladders for inspection of PWR PCCS air baffling and flow passages. In its letter of July 22, 1991, EPRI stated that the BWR PDHR pool, condensing heat exchanger, and associated piping were considered to be vital equipment and would be protected against sabotage. EPRI also identified the PWR PCCS containment steel shell and access to the cooling air intakes and discharges as vital equipment that needed to be protected against sabotage actions. The EPRI letter stated that these components were considered vital as discussed in Section 5.2.1.1 of Chapter 9 of the passive Requirements Document. However, the staff stated in the DSER for Chapter 5 that these components were not explicitly identified as vital equipment in that section. This was identified as an open issue. In its letter dated September 29, 1992, EPRI explained that Section 5.2.1.1 of Chapter 9, rather than providing a specific list of vital equipment, broadly defines vital equipment and requires that a physical security system be provided to protect vital equipment in accordance with 10 CFR 73.55. The staff concludes that, because it specifies that passive plant vendors' designs must meet 10 CFR 73.55, this requirement is acceptable. Therefore, this open issue is closed.

3 ALWR CORE DAMAGE PREVENTION REQUIREMENTS

3.1 Overview

The passive ALWR core damage prevention functions will be provided by the passive safety injection system (PSIS), passive decay heat removal (PDHR) system, control rod drive system, diverse reactivity control system, and depressurization system (DPS). Section 3 of Chapter 5 of the passive Requirements Document specifies the core damage prevention requirements common to both the BWR and PWR designs.

3.2 Passive Safety Injection System

Section 3.2 of Chapter 5 requires that the PSIS provide for passive safety-grade means to supply abundant injection water to the reactor vessel to satisfy the LOCA core cooling limits, assuming no ac power and the most limiting single failure, and that means be provided to permit appropriate periodic surveillance testing of PSIS valves and system components. These ALWR requirements are intended to satisfy the emergency core cooling requirements in GDC 35 and 37 of Appendix A to 10 CFR Part 50. GDC 37 requires the emergency core cooling system (ECCS) to be designed to permit periodic pressure and functional testing. GDC 35 also requires that suitable redundancy in the components and features, and suitable interconnections, leak detection, isolation, and containment capabilities be provided to ensure that system safety function can be accomplished assuming loss of offsite or onsite power and a limiting single failure. Sections 4.8 and 4.35 of Appendix B to Chapter 1 of the passive Requirements Document state that, although the PSIS does not meet either requirements of GDC 35 or the guidance of SRP Section 6.3 that the ECCS function with onsite and offsite power, the ALWR PSIS will function independently of all ac power supplies assuming a single failure; therefore, it complies with the intent of GDC 35 and SRP Section 6.3 by providing abundant capability for core cooling without the necessity of relying on ac power. In response to a staff question, EPRI stated that the PSIS meets the intent of GDC 35 by ensuring that appropriate cladding temperature limits will be satisfied for the full spectrum of break sizes and break locations without any reliance on ac power; therefore, the staff concludes that this is acceptable. However, the design of the PSIS will be reviewed on a plant-specific basis during the staff's review of an individual application for FDA/DC.

A more detailed staff review of the PSIS design requirements is contained in Sections 4 and 5 of this chapter.

Section 3.2 of Chapter 5 requires that the PSIS core coolant inventory be adequate for at least 72 hours without the need for operator action for all postulated licensing design-basis events. The quality of the PSIS makeup water will be as specified in Section 5.5.2.4 (PWR) or Section 5.5.2.1 (BWR) of Chapter 1 of the passive Requirements Document. The staff considers these requirements acceptable. The 72-hour design basis is addressed in Section 2.3.3 of Chapter 1 of this report.

3.3 Passive Decay Heat Removal System - Safe Shutdown

Section 3.3 of Chapter 5 specifies requirements common to the PDHR systems of the BWR and PWR plants. The PDHR systems are required to provide for passive safety-grade means to maintain the core adequately cooled, assuming an initiating event with limiting single failure, and no ac power for 72 hours while the RCS and containment are maintained within design limits for all licensing design-basis events. Requirements for the PDHR systems of the BWR and PWR are specified in Sections 4.3 and 5.3 of Chapter 5, respectively.

Section 3.3.2 of Chapter 5 originally required that plant emergency shutdown and cooldown from reactor critical at full temperature and pressure to a safe stable condition be accomplished with safety-grade PDHR systems within 72 hours (assuming single failure) without relying on ac power. It further specified that a safe stable condition was defined as 400 °F. In the DSER for Chapter 5, the staff stated that EPRI's ALWR position on safe shutdown condition was not consistent with RG 1.139, "Guidance for Residual Heat Removal," and Branch Technical Position (BTP) RSB 5-1 (NUREG-0800), which require plant safe shutdown to be at the cold-shutdown condition.

In Revision 4, EPRI revised Section 3.3.2 of Chapter 5 to require that plant emergency shutdown and cooldown from reactor critical at full temperature and pressure to a safe stable condition be accomplished with safety-grade PDHR systems within 36 hours (assuming single failure) without relying on ac power. It further defines a safe stable condition as 215.6 °C (420 °F). This position on safe shutdown condition is not consistent with RG 1.139 and BTP RSB 5-1, which require plant safe shutdown to be at the cold-shutdown condition using only safety-grade systems. This safe shutdown issue is also discussed in the passive Requirements Document, Section 2.5.6 of Appendix B to Chapter 1, where it is categorized as a plant optimization issue.

Various definitions for a safe shutdown appear in NRC regulations. For example, 10 CFR 50.2 defines "safe shutdown (non-design basis accident)" for station blackout as bringing the plant to those shutdown conditions specified in plant technical specifications as hot standby or hot shutdown, as appropriate. Appendix R to 10 CFR Part 50 states that the phrase "safe shutdown" is used as applying to both hot and cold shutdown. The regulation does not define safe shutdown following normal operation or a design-basis accident (DBA), nor does it define what constitutes a safe shutdown state.

GDC 34 of Appendix A to 10 CFR Part 50 requires that a residual heat removal (RHR) system be provided to remove residual heat from the reactor core so that specified acceptable fuel design limits (SAFDLs) and the design conditions of the reactor coolant pressure boundary are not exceeded. GDC 34 further requires suitable redundancy of the components and features of the RHR system to ensure that the system safety functions can be accomplished, assuming loss of offsite or onsite power, coincident with a single failure. These requirements are intended to ensure availability of the RHR system for long-term cooling to ensure a safe shutdown state.

Because passive ALWR designs use passive heat removal systems for decay heat removal, they are limited by the inherent capability of the passive heat removal processes. These designs cannot reduce the temperature of the RCS below the boiling point of water for the heat to be transferred to the water

pool where heat exchangers are submerged, that is, in-containment refueling water storage tank of the AP600 or the isolation condenser of the SBWR. Even though active shutdown cooling systems are available to bring the reactor to cold-shutdown or refueling conditions, these active RHR systems are not safety grade and do not conform to the guidance of RG 1.139 or BTP RSB 5-1.

The passive Requirements Document states that it is not necessary for the passive safety systems to be capable of achieving cold shutdown based on the belief that the passive ALWR designs meet the GDC 34 requirements by employing a redundant safety system that can operate at full RCS pressure and place the reactor in the long-term cooling modes immediately after shutdown, and maintain fuel and reactor coolant pressure boundary within acceptable limits.

In evaluating EPRI's position with respect to safe shutdown, the staff considered the conditions that constitute a safe shutdown state and assessed the acceptability of EPRI's proposed approach for meeting GDC 34. The staff's position in RG 1.139 and BTP RSB 5-1 to bring the plant to cold-shutdown conditions was based on the desire to cool the RCS enough so that plant inspection and repair could be performed. There are other plant conditions that can constitute a safe shutdown state as long as reactor subcriticality, decay heat removal, and radioactivity containment are properly maintained on a long-term basis.

The passive Requirements Document specifies performance requirements for the PDHR systems to have sufficient capacity to reduce reactor coolant temperature to 215.6 °C (420 °F) within 36 hours of reactor shutdown. To ensure conformance with GDC 34 requirements, the passive Requirements Document also specifies that safety-grade decay heat removal from the RCS will be possible from full RCS operating pressures and temperatures to a safe stable condition for all plant conditions, assuming a single failure. Additionally, EPRI requires that operation of the plant in the long-term cooling mode be automatic, eliminating operator actions to cool down the plant. Also, operation of the PDHR system does not require any ac power, pumps, or valve operation after initial alignment, or support systems (such as component cooling water or service water), and is stable and self-contained, requiring no makeup water for a period of at least 3 days following reactor shutdown. Therefore, a safe stable condition can be maintained with the safety-grade PDHR system.

In addition, after initial cooldown by the PDHR system or main steam system, a non-safety-grade reactor shutdown cooling system will be available to bring the plant to cold-shutdown conditions for inspection and repair. EPRI states that these non-safety-grade systems are required to be highly reliable, and there is no single failure of these systems or their support systems that would result in the inability to terminate use of the passive safety-grade system and achieve cold shutdown, if desired.

The staff believes that the PDHR systems offer potential advantages over current active systems, and can maintain the plant in conditions that are fully consistent with the requirement of GDC 34 to maintain the fuel and reactor coolant pressure boundary within acceptable limits and, therefore, contain radioactivity. In addition, protection against loss of reactor coolant inventory during long-term PDHR operation is available, if necessary,

from the PSIS and the associated depressurization system. These passive system capabilities must be demonstrated by appropriate evaluations during detailed design analyses, including the following:

- A safety analysis, assuming the limiting single failure, to demonstrate that the capabilities of the passive systems are such that they can bring the plant to and maintain a safe stable condition, that no transients will result in the SAFDLs and pressure boundary design limit being violated, and that, with operation of other passive safety systems, such as the PSIS and DPS, no potential high-energy piping failure being initiated from this condition will result in violation of 10 CFR 50.46 criteria.

- A probabilistic reliability analysis including events initiated from the safe shutdown conditions to demonstrate meeting the safety goals. The PRA also determines the reliability/availability missions of risk-significant systems and components as a part of effort pertaining to RTNSS.

However, the staff is concerned that, with the passive system design basis of 72-hour capability, the passive RHR system water pool, without refill, will have water capacity to permit only 72 hours of operation after a scram. A long-term safe stable condition can be maintained if a reliable non-safety support system or equipment is available to replenish the water pool to sustain long-term operation of the PDHR system after 72 hours. The passive Requirements Document requires that non-safety equipment necessary for plant recovery after the assumed 72-hour accident duration be designed for the expected environment, and that only simple, unambiguous operator actions and easily accomplished offsite assistance be necessary after 72 hours to prevent fuel damage. As discussed in Section 2.2 of Chapter 3 of this report, EPRI has proposed an acceptable process for RTNSS. Any non-safety systems that are required for maintaining long-term stable shutdown conditions will be included as part of the RTNSS evaluation and, if found risk significant, will be subject to proper regulatory oversight commensurate with their reliability/availability missions and risk significance. The staff expects that the non-safety support systems and equipment will be evaluated as a part of this process to ensure appropriate design and reliability/availability to provide backup to passive systems beyond 72 hours. This will ensure proper operation of the passive RHR system to maintain a safe stable condition on a long-term basis, as well as reliable non-safety systems that will be necessary to bring the plant to cold shutdown conditions.

The staff concludes that cold shutdown is not the only safe stable shutdown condition to maintain the fuel and reactor coolant boundary within acceptable limits. It concludes that the EPRI-proposed 215.6 °C (420 °F), rather than the cold-shutdown condition required by RG 1.139, as a safe stable shutdown condition that the passive DHR systems must be capable of achieving and maintaining following non-LOCA events, is acceptable. This acceptability is based on an adequate passive safety system performance and appropriate RTNSS evaluations and regulatory oversight measures to ensure availability of the risk-significant non-safety systems that are required for maintaining long-term stable conditions. During its review of an individual application for FDA/DC, the staff will review relevant evaluations described above to verify the safe shutdown condition.

The staff's evaluation of EPRI's proposed 215.6 °C (420 °F) as a safe shutdown condition is also described in Section C, "Safe Shutdown Requirements," of SECY-94-084 in Annex B of Appendix B to Chapter 1 of this report. The staff's position safe shutdown was approved in the Commission's SRM dated June 30, 1994, which is also provided in Annex B of Appendix B to this report.

Section 3.3.4 of Chapter 5 originally specified that the PDHR heat exchangers meet the following requirements:

- The tubing will be fabricated of austenitic stainless steel. The tubes will be expanded over the full thickness of the tube sheet and will be welded to the tube sheets or to the inlet and outlet headers by full penetration welds.

- The heat exchanges will have provisions for cleaning the tubes.

- An appropriate margin in heat transfer area will be included in the design.

- Access to the heat exchanger tubes will be provided for inservice inspection.

- The PDHR heat exchangers will be considered "critical" components in accordance with Chapter 1, Section 5.2.2, and all the materials used, the major criteria for selecting the materials, and any special requirements will be identified.

In the DSER for Chapter 5, the staff stated that these requirements were acceptable because stainless steel tubing would be used in the construction of the PDHR heat exchangers thus minimizing crevice corrosion. Also, access would be provided for inservice inspection. However, the staff concluded that Section 3.3.4 should be clarified to require that low carbon austenitic stainless steel be used for the PDHR heat exchangers. In addition, the piping material for the PDHRS should be selected on the basis of expected environment and operating conditions. This was identified as an open issue. In its letter dated September 29, 1992, EPRI clarified that the generic materials requirements of Section 5.3.1.8 of Chapter 1 apply to the materials selection for the PDHR heat exchangers. The staff concludes that this is acceptable; therefore, this open issue is closed.

3.4 Diverse Reactivity Control System

GDC 26 requires that two independent reactor control systems based on different design principles be provided: one of the systems will use the control rods and the second will be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to ensure fuel design limits are not exceeded.

Section 3.5 of Chapter 5 requires a diverse reactivity control system to be provided that meets the applicable requirements of GDC 26. It indicates that the standby liquid control system will be used to provide the diverse reactivity control function for the BWR, and the electric motor drive of the electrohydraulic control rod drive will be used as the backup scram mechanisms to satisfy ATWS requirements. For PWRs, the chemical and volume control system

(CVCS) is not required to be a safety-grade system. The diverse reactivity control function is provided by the PSIS, which provides boration of the RCS for accident mitigation or achieving safe shutdown as described in Section 5.2. The staff concludes that these requirements do not conflict with regulatory guidance or Commission regulations.

Although EPRI does not explicitly address ATWS in Chapter 5 of the passive Requirements Document, the requirements in Section 3.5, discussed above, are in compliance with 10 CFR 50.62 (referred to as the ATWS rule). The staff concludes that compliance with the ATWS rule is an acceptable approach for resolving the issue, except in the area of diversity. The staff noted that when the ATWS rule was implemented, certain PWR facilities, whose design includes features that would reduce the consequences of an ATWS, were not required to backfit a diverse scram capability. However, a diverse scram capability is an important measure to help prevent core damage for all ALWRs, especially when incorporated into the initial design. Therefore, in the DSER for Chapter 5, the staff concluded that the passive Requirements Document should include a requirement that, unless the ALWR designer can demonstrate that the consequences of an ATWS are acceptable, the ALWR PWR have diverse reactor protection system (RPS) input to the control rods. This was identified as an open issue. In its letter dated September 29, 1992, EPRI explained that Section 8 of Chapter 10 provides specific requirements for diverse RPS input unless the designer can demonstrate that the plant will withstand an ATWS event. As described in greater detail in Section 8 of Chapter 10 of this report, the staff concludes that this is acceptable. Therefore, this open issue is closed.

4 BWR CORE DAMAGE PREVENTION REQUIREMENTS

Section 4 of Chapter 5 of the passive Requirements Document specifies the performance and equipment requirements for the BWR safety-grade core damage prevention systems, that is, the passive safety injection system (PSIS), the passive decay heat removal (PDHR) system, the depressurization system (DPS), and the standby liquid control system (SLCS). The core coolant inventory control functions will be carried out by the PSIS and PDHR in conjunction with the DPS; the decay heat removal function will be carried out by the PDHR system; and the diverse reactivity control function will be provided by the SLCS.

The BWR also will use several non-safety grade systems to achieve decay heat removal during normal plant shutdown and refueling. These include the normal feedwater system and steam flow to the main condenser, the control rod drive pump in conjunction with the steam relief valves to provide backup reactor feed, and non-safety PDHR systems for decay heat removal during normal cooldown and refueling.

4.1 Elimination of Core Spray

The passive Requirements Document does not require that the design for a BWR include a core spray system because there will be no large recirculating system piping and, therefore, no postulated large or medium loss-of-coolant accidents (LOCAs) below the core. In addition, reflood cooling will be sufficient to protect against core damage. The staff concludes that this is acceptable if detailed LOCA calculations confirm that core spray is not necessary to meet 10 CFR 50.46 requirements. In the DSER for Chapter 5, the staff stated that it would require the ALWR designer/applicant to perform and demonstrate the acceptability of such an analysis. In its letter dated September 29, 1992, EPRI agreed that it is appropriate for the staff to verify that specific plant designs meet 10 CFR 50.46 analysis requirements to demonstrate that there is no need for a core spray system. Therefore, the staff will evaluate the LOCA calculations during its review of an individual application for FDA/DC to ensure that the requirements of 10 CFR 50.46 are met, with margin, using the safety-related equipment that is provided.

4.2 BWR Passive Safety Injection System

The PSIS is a safety system that will consist of a PSI water storage pool open to the containment atmosphere, water supply, piping and valves between the PSI pool and the reactor vessel, and instrumentation and controls. The PSIS will provide core coolant inventory makeup for LOCA events if the non-safety systems are not available or are not adequate. The system will work in conjunction with the DPS, which will depressurize the reactor, allowing the injection of core coolant inventory makeup by gravity.

Section 4.2.3 of Chapter 5 specifies the PSIS and equipment requirements. Section 4.2.3.1.1 specifies that the PSIS will be divided into an appropriate number of redundant components and features (e.g., multiple piping for core coolant makeup from the PSIS pool and suppression pool and redundancy of valves, controls, and instrumentation). EPRI also specifies that a single PSIS pool may be used and that the total pool volume may be provided by a

number of segments that are connected in such a way that they perform as a single unit. In response to a staff concern that a line break might affect more than one subsystem if all subsystems shared a common pool, EPRI stated that the pool would be segmented but connected to act as one unit, and the plant designer would be required to provide redundancy and separation in accordance with regulatory requirements to ensure the safety function is performed after an initiating event and a single failure. The staff finds EPRI's response to be acceptable. It will evaluate the redundancy and separation requirements for the PSIS pool design during its review of an individual application for FDA/DC.

Section 4.2.3.1.3 of Chapter 5 requires each of the core coolant injection lines of the PSIS to be connected directly to the reactor vessel and to enter into the annulus between the core and the vessel with the elevation of the nozzles being above the top of the fuel. EPRI indicates that this arrangement will enhance overall system reliability by ensuring that a line break will not affect more than one subsystem. The staff expressed concern that the lack of specific requirements regarding the PSIS injection arrangement to prevent condensation-induced water hammer could result in pressure disturbances and affect system operation. In response, EPRI stated that water hammer should not be a concern because the PSIS injection lines will be under water on the high-pressure side and will be at containment pressure and full of water on the pool side. The squib valves, which will be the high/low-pressure interface, will be designed to be zero leakage valves, thereby preventing high-pressure and high-temperature water from leaking to the low-pressure side and causing steam pockets. The only time that flow will occur in the injection lines will be during low-pressure conditions; thus, there should be no concern regarding water hammer. In addition, Section 4.5.4.1.3 of Chapter 1 requires the plant designer to consider potential water hammer events such as turbine stop valve closure, safety/relief valve actuation, fast valve actuations, and rupture disc loads in the design of plant systems.

The staff notes that condensation water hammer is caused by the formation of steam pockets so that a sudden condensation of the steam in contact with subcooled water slug results in instantaneously localized depressurization with acceleration of the water slug toward the voids. Therefore, the conditions conducive to a damaging water hammer require not only the subcooled water, but also a system arrangement where an abundance of entrapped steam pockets can be formed by the water slugs. Since the passive Requirements Document does not provide a detailed design arrangement, the staff concludes that condensation water hammer is beyond the scope of the passive Requirements Document. Therefore, it will evaluate the potential for condensation water hammer during its review of an individual application for FDA/DC.

Sections 4.2.3.1.2 and 4.2.3.2.2 of Chapter 5 require the PSIS water pool, in conjunction with the suppression pool, to provide injection flow into the reactor vessel entirely by gravity and to have an adequate amount of water stored to ensure the core is covered with water for all licensing design-basis accidents. Section 4.2.3.2.3 requires the PSIS pool to be located in the upper portion of the containment, sufficiently high to provide the driving head for injection of the core coolant makeup into the reactor vessel. Even though Section 4.2.3.1.2 states that the suppression pool also will be required for the core coolant makeup by gravity, there is no specific requirement for the driving head of the suppression pool. In response to a staff

question, EPRI stated that the capability for suppression pool inventory injection into the reactor vessel requires only a minimal driving head. If suppression pool water is needed for RCS makeup, the reactor will be fully depressurized and the open line between the suppression pool and the reactor vessel will allow the water level in the reactor vessel to be equalized with the suppression pool; therefore, a large driving head is not required. This explanation is acceptable; however, the staff will evaluate the plant-specific design to ensure sufficient suppression pool driving head is available during its review of an individual application for FDA/DC.

4.3 BWR Passive Decay Heat Removal System

The PDHR system will be safety related and designed to remove reactor decay heat during loss of all onsite ac power or when the non-safety DHR systems are not available. The system will be designed to remove heat by condensation of steam from the reactor vessel at any system pressure, from full operating pressure to PSIS operating pressure. The PDHR system will consist of condensing heat exchangers located in a water storage pool and related supply and return piping and valve arrangements, as well as necessary instrumentation and controls.

Section 4.3.2 of Chapter 5 originally specified the performance requirements for the PDHR system. It required the system (1) to have sufficient heat removal capacity to prevent safety/relief valves from opening after the main steam isolation valves (MSIVs) closed while operating at full reactor power; (2) to be capable of continued DHR during and after a small-break LOCA up to the size that causes reactor water level to fall to the automatic depressurization set point; (3) to be capable of removing heat from the reactor vessel and from the containment following a LOCA and maintaining the containment pressure within design limits; (4) to be capable of removing decay heat and maintaining the reactor at pressure during hot standby; and (5) to be initiated automatically for all RCS isolation events that close the MSIVs when the reactor is in the "run" mode.

The capability of the PDHR system to remove heat from the reactor vessel and from the containment following a LOCA and to maintain the containment pressure within design limits was intended to avoid the need for another system solely for containment heat removal. However, in Revision 4, this capability requirement was removed. A requirement was added for a separate passive containment cooling system for the BWR design to remove reactor decay heat from the containment following a LOCA or a reactor depressurization event, as specified by Section 7.4 of Chapter 5.

In the DSER for Chapter 5, the staff stated that there appeared to be an inconsistency between the requirements for the PDHR system to be capable of removing the decay heat during hot standby and to be capable of achieving and maintaining the safe shutdown condition at 400 °F specified in Section 3.3.2 of Chapter 5. The staff concluded that EPRI should clarify its intent for defining the safe shutdown condition as hot standby or 400 °F and identified this as an open issue.

As discussed by the staff in Section 3.3 of Chapter 5 of this report, Section 3.3.2 was revised in Revision 3 to define a safe stable condition at 215.6 °C (420 °F), and to require plant emergency shutdown and cooldown with

safety-grade decay heat removal systems from reactor critical at full temperature and pressure to a safe stable condition within 36 hours without relying on ac power. The PDHR system is required to be able to remove decay heat at any system pressure. It can, therefore, be used to remove heat during operation in hot standby including heat removal down to the safe shutdown state. The staff concludes that EPRI's safe shutdown definition has been clarified. Therefore, this open issue is closed.

The staff has determined that EPRI's requirements do not completely conform to the guidance of RG 1.139 to bring the reactor to a cold-shutdown condition, which is a safe shutdown condition, within 36 hours of reactor shutdown. However, the passive Requirements Document requirement pertaining to the PDHR system capacity to reduce the reactor coolant temperature to 420 °F within 36 hours of reactor shutdown is consistent with the intent of RG 1.139 to maintain the reactor in a stable, subcritical condition, with adequate cooling to prevent fuel cladding failure, as long as this end state can be maintained as a safe shutdown condition. The staff evaluates the adequacy of this end state for possible plant safe shutdown in Section 3.3 of this chapter. It concludes that 420 °F is an acceptable safe stable shutdown condition, provided there are acceptable passive safety system performance and appropriate regulatory oversight measures to ensure the availability of risk-significant non-safety systems that are required for maintaining long-term stable shutdown conditions.

Section 4.3.3 of Chapter 5 specifies BWR PDHR system and equipment requirements. These include requiring a suitable number of redundant components and features (Section 4.3.3.1), including redundant injection valves on each of the condensate return lines to the reactor vessel (4.3.3.7) to meet the single-failure criteria or redundancy requirement of GDC 34 for decay heat removal systems, and requiring that each condensing heat exchanger be located in a water pool sufficiently high to provide for gravity return of condensate to the reactor vessel (4.3.3.2).

Section 4.3.3.3 of Chapter 5, permits the use of a single water pool with suitable separation of heat exchangers and piping. The rationale portion indicates that mechanical separation of the heat exchangers and piping may be required to protect against common-mode failure. The staff concludes that this explanation is acceptable, but recommends that EPRI clarify this requirement to ensure that the plant designer considers protection against common-mode failure.

Section 4.3.3.4 of Chapter 5, requires that the water pool be located outside the primary containment but inside a structure adequate to provide the physical protection required for a safety-related system and that the pool be adequately vented to the atmosphere to prevent overpressurization of the pool structure. Because the water pool will be vented to the atmosphere, the staff was concerned that a breach of a heat exchanger tube could result in a LOCA outside the containment. The staff evaluation of this concern is given in this report in Section 2.5 of this chapter.

Section 4.3.3.5 of Chapter 5, requires sufficient water capacity to be available in the PDHR water pool to permit 72 hours of operation after scram without the need to refill the pool. The staff concludes that the 72-hour PDHR water pool capacity requirement is acceptable, provided proper means are

available to replenish the pool water for extended operation of the PDHR system after 72 hours. In Revision 3 of the passive Requirements Document, Section 2.2.10 of Chapter 5 requires that the passive safety systems continue to be available for an extended period to perform their safety function by use of onsite facilities and by connecting readily available offsite equipment and resources to replenish these inventories and provide support, as may be necessary. As discussed in Section 2.3.2 of Chapter 1 of this report, the 72-hour design basis and operator actions 72 hours after accidents will be evaluated as an integral part of the process for evaluating the RTNSS. The results of the design-specific evaluation of the RTNSS review will determine the risk significance and reliability/availability mission of the non-safety systems (such as onsite facilities and offsite equipment and resources) needed to replenish the water pool after 72 hours. Proper regulatory oversight of these non-safety systems will then be developed to ensure adequate system functionality and reliability.

The passive Requirements Document does not specify whether the heat exchanger tubes should remain covered throughout the 72-hour period, nor does it specify consideration of the PDHR pool level reduction caused by boiloff in the 72-hour pool water capacity for all accident scenarios. In the DSER for Chapter 5, the staff identified this as an open issue. In its letter dated September 29, 1992, EPRI clarified that because the intent of the passive Requirements Document is to ensure that the heat removal function is performed without the need for operator action for at least 72 hours, plant designers will have to consider the PDHR pool level reduction caused by boiloff in the 72-hour pool water capacity for all accident scenarios and must demonstrate with licensing design-basis analyses that the heat exchangers perform their function for the time they are needed, including reduction in heat transfer capability as a result of level reduction that might uncover the heat exchanger tubes. EPRI also requires that designers provide test or experience data to support the functionality of the PDHR system in accordance with the requirements of Section 1.2 of Chapter 1, which specifies that only proven technology will be used. The staff concludes that EPRI's requirement for plant designers to demonstrate the adequacy of their designs to provide PDHR for 72 hours without operator action is acceptable. The staff will review individual applications for FDA/DC to ensure that designers provide licensing design-basis analyses to demonstrate the adequacy of the PDHR heat exchangers to perform their intended function in all accident scenarios for at least 72 hours without operator action needed to restore PDHR pool level. Therefore, this open issue is closed.

Section 4.3.3.6 of Chapter 5 specifies that reactor steam and condensate will flow inside the tube of the PDHR system heat exchanger, that vents will be provided to remove noncondensibles during standby or during operation, and that the standby vent will discharge to the main condenser while the operation vent discharges to the suppression pool. EPRI does not provide guidance regarding the location of vent connections and does not consider the effects of the ingress of any residual (unvented) noncondensibles on heat transfer of condensing heat exchangers, pressure drop, and natural circulation. In response to a staff question, EPRI indicated these were beyond the scope of the passive Requirements Document and would be determined by the plant designer. The staff will evaluate these items during the review of an individual application for FDA/DC.

Section 4.3.3.8 originally required that the injection actuation valves open in the event of loss of control or motive power to ensure that a failure would not prevent PDHR system operation. However, in Revision 4 of the passive Requirements Document, this section was revised to specify that one of the condensate return valves (or a second valve in parallel) must be a throttling valve suitable for controlling reactor pressure. Section 4.3.3.9 was added to require that means be provided for operator control (from the main control room) of both the injection valve and throttling valve. These requirements provide means to allow the operator to maintain reactor system pressure under various post-isolation conditions. Similar requirements for the operator to control the RCS cooldown rate are also specified in Section 5.3.2.3 of Chapter 5 for the PDHR system for the PWR designs.

In the DSER for Chapter 5, the staff stated that it was concerned that in using modulating valves in the PDHR systems to control the cooldown rate, operator error, or control system failure could defeat the safety function of PDHR. In its letter dated September 29, 1992, EPRI stated that the specific manual actions have yet to be defined; however, any manual action, such as throttling the flow rate or tripping the reactor coolant pumps to reduce the flow rate to that achieved by natural circulation, would be in accordance with operating procedures. As no detailed information is available, the staff will evaluate the appropriateness of the PDHR system throttling design, operating procedures, and operator action to control the reactor system pressure during its review of an individual application for FDA/DC.

Section 4.3.3.12 of Chapter 5 requires the PDHR steamline to the heat exchangers to be arranged to slope down to the reactor vessel without pockets and the high point to be at the heat exchangers. EPRI indicates, and the staff agrees, that this requirement is necessary to avoid buildup of condensed steam in the system piping and the potential for water hammer and other deleterious effects of pockets of condensed steam.

4.4 BWR Depressurization System

The DPS is a safety-related system designed to provide reactor depressurization capability if that passive safety injection is required. The system will include the instrumentation and controls for operation of the safety/relief valves (SRVs) that will be used for the DPS functions and of the depressurization valves (DPVs).

Section 4.4.1.2 of Chapter 5 specifies that actuation of the DPS will provide a potential pathway through which the drywell atmosphere may flow backward through the open valves into the reactor steam space and consequently into the PDHR system to provide containment cooling (Section 4.3.1.2 of Chapter 5 originally required the BWR PDHR system to be designed to remove decay heat from the containment following a LOCA). In reviewing the arrangement of the PDHR and containment cooling system in Figure 5.4-2, the staff questioned if the DPS valves will be the only pathway for containment cooling by the PDHR system and if the PDHR containment cooling will be operable only when the drywell pressure is higher than the RCS pressure. The staff also questioned the specific EPRI guidelines regarding the piping and valve arrangements and the control system for DHR from both the reactor vessel and the containment, considering a common-mode failure of the DPS may prevent containment heat removal. EPRI responded that Figure 5.4-2 was only a sample arrangement and

that a vendor design team was evaluating a design that used an additional set of isolation condensers for the containment cooling function. This design would leave the containment cooling heat exchangers continuously open to the drywell atmosphere, thereby eliminating the need for any valves and eliminating the potential for a DPS common-mode failure affecting containment cooling.

In Revision 4 of the passive Requirements Document, Section 7.4 of Chapter 5 was revised to require a passive containment cooling system that is separated from the PDHR system for the BWR designs, and Section 4.3 was revised by deleting the functional requirement that the PDHR system in the BWR designs be designed to remove decay heat from the containment following a LOCA. Therefore, the staff is no longer concerned that a common-mode failure of the DPS may prevent containment heat removal.

Section 4.1.2.2 of Chapter 5 requires the BWR DPS valves to be opened in stages. Section 4.4.2 requires that the DPS be capable of depressurizing the reactor vessel to the extent that drywell and reactor pressure reach equilibrium and that the depressurization rate not be excessively rapid but still ensure PSIS injection before the core is uncovered. Section 4.4.3 requires that the DPS work in conjunction with the PSIS and have the same system initiating signals and that the DPS valves work in conjunction with the SRVs and ATWS initiation signals to provide additional reactor vessel relief capacity if needed during as ATWS event. EPRI states that the plant designer will determine the specific number of valves and the stages for depressurization based on the detailed system design and the functional requirements in the passive Requirements Document. The staff concludes that these requirements do not conflict with Commission regulations; however, it will perform a further evaluation during its review of an individual application for FDA/DC.

SECY-93-087 gives the staff position on DPS reliability. As part of severe-accident mitigation and to reduce the potential for containment failure as a result of direct containment heating from high-pressure core debris, the Commission paper provides the general criterion that the passive plant design must include a reliable DPS. The passive Requirements Document includes requirements that the DPS have sufficient redundancy so that the safety performance requirements of the system can be met given a single failure (4.4.3.2). In addition, the passive BWR is required to have DPS actuation circuitry that is testable at power (4.4.3.5), and the actuating devices must be testable in place or easily removed for testing (4.4.3.6). These requirements provide for regular surveillance testing to ensure that the system is operational. The passive BWR design also provides diversity since the SRVs and the DPVs both serve depressurization functions and are required to be of independent designs in order to minimize the vulnerability to a common-mode failure (4.4.3.7). These requirements provide adequate assurance that the DPS will be reliable; therefore, the staff finds this acceptable to meet the SECY-93-087 staff position on DPS reliability.

4.5 BWR Standby Liquid Control System

The standby liquid control system (SLCS) will consist of a high-pressure accumulator tank containing a liquid control chemical, piping, valves, and controls and instrumentation. Section 4.5 of Chapter 5 states that the SLCS's functions will be to provide a means of inserting negative reactivity into the reactor core separate and diverse from the control rod system and to provide

the capability for reactor shutdown from full-power operation to cold subcritical with no assistance from control rod movement. The SLCS will be used if a sufficient number of control rods cannot be inserted to maintain subcriticality, but it will not be required to shut down the reactor or to be a backup system for the control rod drives.

Section 4.5.2.4 of Chapter 5 requires that the SLCS be initiated by the operator, be capable of maintaining a coolable fuel geometry, and be capable of maintaining the reactor vessel and containment below their design limits assuming failure of normal scram and backup electric motor drive while at normal operation. The passive Requirements Document does not specify the maximum allowable time for an operator to initiate the SLCS. In response to a staff question, EPRI indicated that preliminary indications were that the operator will have at least 30 minutes to initiate the SLCS without exceeding the reactor pressure vessel (RPV) overpressurization limit, but that it was the responsibility of the plant designer to perform the analysis to support the time an operator has for manual initiation of the SLCS. However, as noted below, the staff requires automatic initiation of the SLCS consistent with 10 CFR 50.62.

The ATWS rule (10 CFR 50.62) requires that each BWR (1) have an alternate rod injection (ARI) system that is diverse and independent of the reactor trip system, (2) have an automatically initiated SLCS with the capability of injecting a borated water solution with the resulting reactivity control at least equivalent to 86 gpm of 13-weight percent sodium pentaborate decahydrate solution at the natural boron-10 isotope enrichment into an RPV with an inside diameter of 251 inches, and (3) have equipment to automatically trip the reactor coolant recirculation pumps under conditions indicative of an ATWS.

Since the passive BWR will operate using natural circulation as the coolant motive force, the requirement for an automatic reactor coolant recirculation pump trip is not applicable. As indicated in Chapter 3 of this report, the passive BWR will use automatic feedwater runback to lower vessel level and reduce reactivity. Also, as specified in Section 5.3.6 of Chapter 4 of the passive Requirements Document, the passive BWR control rod drive system will include a provision for an alternate means of rod insertion, using a means separate and diverse from the normal scram by the reactor trip system. Therefore, the only deviation from the ATWS rule is the manual initiation of the SLCS. Section 2.5.5 of Appendix B to Chapter 1 of the passive Requirements Document originally identified the manual initiation of the SLCS as a plant optimization subject and concluded that the need for an automatic SLCS was negated because of enhanced reliability of the hydraulic/electric motor-driven control rod system that could be initiated by the reactor protection system and alternate rod insertion system.

In response to staff questions regarding ATWS analyses to justify manual SLCS initiation, EPRI indicated that ATWS events analyses were the responsibility of the plant designer and were beyond the scope of the passive Requirements Document. However, EPRI also stated that a sufficient redundant and diverse reactivity insertion capability would be provided and that the need for automatic SLCS actuation would be determined by reliability and accident response analyses performed by the plant designer. If necessary, the plant designer would require automatic SLCS initiation. In the DSER for Chapter 5, the staff concluded that manual SLCS was not acceptable because EPRI had not

provided an acceptable basis. In Revision 4, EPRI made several revisions related to automatic actuation of SLCS. Section 4.5.2.4 has been revised by deleting operator initiation from the system injection requirement. Section 4.5.3.5.1 of Chapter 5 and Section 8.7.2 of Chapter 10 have been revised by requiring that the SLCS be automatically actuated on signal(s) that are unique to an ATWS event, that the automatic actuation logic be highly reliable and designed to avoid inadvertent operation, and that capability of manual initiation by direct operator action in the main control room be provided. In addition, Section 2.5.5 of Appendix B to Chapter 1 has been deleted as a plant optimization subject. The staff concludes that, because they now require automatic initiation of the SLCS, these revisions comply with the requirements of 10 CFR Part 62 and are acceptable.

The staff concludes that the 10 CFR 50.62 requirements for ATWS mitigation are very prescriptive for the current generation of BWRs and were based on insights drawn from analyses of current designs. To show that these insights are still applicable, the staff will require ATWS analyses for the passive BWR. The intent of these analyses is to demonstrate that the assumptions leading to 10 CFR 50.62 requirements are still valid. These analyses should include the effects of natural circulation, stability, hydraulic scram failure, and slow reactivity insertion by the electric fine-motion rod control and the effects on thermal limits at the top of the core where there may be high power peaking when motor-driven scram is used.

Section 4.5.2.6 of Chapter 5 requires the adequacy of boron mixing and distribution to be determined by analyzing test results to demonstrate that the liquid control does not recirculate through the core in uneven concentrations, which would cause an appreciable rise and fall in reactor power or the settling of higher concentrations in non-core regions. Although the passive Requirements Document states that tests have been run on various BWR configurations to provide a basis for mixing and distribution analyses, EPRI states that the determination and specification of the necessary minimum flow rate is beyond the scope of the passive Requirements Document and is the responsibility of the plant designer. Therefore, the determination of the minimum flow rate, as well as the associated tests, will be evaluated during the staff's review of an individual application for FDA/DC.

Section 4.5.2.7 of Chapter 5 specifies that the SLCS liquid control will be a 10 percent by weight aqueous solution of sodium pentaborate and that the solution temperature will be maintained at 5 °F or more above the solution saturation temperature during plant operation by the normal heating, ventilating, and air conditioning (HVAC) system. The passive Requirements Document indicates that the 10 percent solution, which has a saturation temperature of approximately 38 °F, avoids the need for special heating. The staff requested an analysis to demonstrate that this ALWR specific sodium pentaborate concentration meets of 10 CFR 50.62, which requires that the BWR SLCS be capable of providing a reactivity control equivalent to that resulting from injecting 86 gpm of 13-weight percent sodium pentaborate decahydrate solution into an RPV with an inside diameter of 251 inches for a given core design. In response, EPRI stated that the analysis was beyond the scope of the passive Requirements Document and the responsibility of the plant designer. This will be evaluated during the staff's review of an individual application for FDA/DC.

In the DSER for Chapter 5, the staff stated that Section 4.5.3.1.1 of Chapter 5 required the use of a high-pressure accumulator for SLCS injection and storage, whereas Section 4.5.3.1.2 specified a gravity drain tank inside the containment with top and bottom connections to the RCS to be considered as an alternative. The passive Requirements Document indicated that a gravity drain tank concept might be attractive because it would eliminate the need for containment penetrations and complexities associated with a charging system; however, it provided no specific requirements or guidelines for the design of a gravity drain tank. In response to a staff question, EPRI stated that the gravity drain design was not being pursued at that time and no specific requirements had been developed. Therefore, the staff concluded in the DSER for Chapter 5 that a gravity drain tank concept for the SLCS was unacceptable and that this option should be removed from the passive Requirements Document for clarity. This was identified as an open issue. In Revision 4, Section 4.5.3.1.2 and the option for a gravity drain tank have been deleted. This resolves the staff's concern and is acceptable. Therefore, this open issue is closed.

Section 4.5.3.1.3 of Chapter 5 requires the accumulators to be equipped with a float device at the injection flow exit to shut off flow when level has dropped, thus preventing unnecessary flow of accumulator charge gas into the reactor vessel at the end of the injection period. However, no requirement is specified for the shutoff level of the float device to ensure that an adequate volume of poison will be injected into the RCS before the flow is shut off. The staff is concerned that a float device malfunction may result in shutting off the flow before sufficient poison has been injected. EPRI indicated that these concerns were the responsibility of the plant designer. This will be evaluated during the staff's review of an individual application for FDA/DC.

Section 4.5.3.2.1 of Section 5 requires the SLCS accumulator to be maintained by a charge of nitrogen gas at a pressure sufficient to meet the performance requirements. The staff asked EPRI if the accumulator pressure requirement considered the accumulator pressure decrease as a result of injection and of RCS pressure increase during ATWS events. EPRI indicated that this was beyond the scope of passive Requirements Document and was the responsibility of the plant designer. The staff will review the plant-specific analysis to demonstrate adequacy of the accumulator pressure for the SLCS injection during ATWS events during its review of an individual application for FDA/DC.

5 PWR CORE DAMAGE PREVENTION REQUIREMENTS

Chapter 5 of the passive Requirements Document specifies the performance and equipment requirements for the PWR safety-grade core damage prevention systems. These systems will include the passive safety injection system (PSIS), the passive decay heat removal (PDHR) system, and the depressurization system (DPS).

5.1 System Functions

The primary function of the DPS will be to depressurize the reactor coolant system (RCS) when inadequate RCS inventory exists so that PSIS injection can occur. The PSIS, operating in conjunction with the DPS, will carry out the core cooling inventory control function when RCS inventory cannot be maintained by the normal charging pumps.

The PDHR system will perform the following safety-related DHR functions: (1) DHR when no feedwater is available to the steam generator, (2) safe shutdown, and (3) backup to the non-safety-grade reactor shutdown cooling system if that system were to become unavailable during normal cooldown.

5.2 PWR Passive Safety Injection System

The PSIS will be a dedicated safety system designed to perform the following functions: provide coolant makeup to the RCS following a loss-of-coolant accident (LOCA), inject sufficient boron into the RCS for safe shutdown, inject boron into the RCS following a main streamline break for supplemental reactivity control, and prevent excessive boron concentration in the RCS in the long term following a cold-leg LOCA. The PSIS will include the in-containment refueling water storage tank (IRWST), accumulators, core makeup tank (CMT), and associated valves, piping, controls, and instrumentation. During high-pressure scenarios, such as small-break LOCAs, the CMTs will inject by gravity since they will be pressure balanced to the RCS and pressurizer (at full RCS pressure). As CMTs empty, the DPS depressurization valves (DPVs) will open automatically in stages to depressurize the RCS to the point at which water can drain by gravity from the low-pressure IRWST into the RCS. Injection flow from the CMTs will be supplemented during large-break LOCAs by the accumulators and IRWST. The accumulators will supply water to the RCS once the RCS pressure is reduced to below 700 psig by action of the DPS; the IRWST will inject when the RCS pressure decreases to within a few pounds per square inch of that of the containment.

Since the PWR PSIS will inject borated water from the CMTs and accumulators into the reactor core by natural circulation, mixing of the highly borated water with the reactor coolant may be relatively inefficient compared to that at plants with forced safety injection systems. In addition, depressurization by the actuation of the automatic depressurization system will result in low RCS pressure and a saturated two-phase fluid system with a high density ratio. The two-phase fluid may collapse into single phase caused by pressure waves induced by cycling relief valves, resulting in a large variation in core coolant density. These phenomena may invalidate the appropriateness of the point kinetics models used in the current PWRs for reactivity and power

calculations. Even though Section 2.5 of Chapter 1 of the passive Requirements Document specifies the requirement that NRC-approved methods be used for analysis of the licensing design-basis events, the staff questioned the EPRI guidelines regarding the use of proper kinetics models for reactivity and power profile calculations and other models to track boron wave movement, diffusion and mixing, and other important thermal-hydraulic phenomena, including condensation in equalization lines, natural circulation, and recirculation from the containment sump. In response, EPRI stated that the designer will have to identify phenomena unique to the passive plant and perform the necessary work to demonstrate that the models adequately account for these phenomena. EPRI indicated that it was beyond the scope of the passive Requirements Document to provide specific guidelines for each unique phenomenon that might be identified. Therefore, the staff will evaluate this during the review of an individual application for FDA/DC.

Figure 5.5-1 of Chapter 5 depicts the PWR PSIS arrangement, and Section 5.2.3 of Chapter 5 specifies the requirements for the PSIS and equipment.

Section 5.2.3.4 of Chapter 5, which specifies the equipment requirements for the CMTs, does not specify requirements for any pressure relief capability for the CMT. The staff is concerned whether there is capability to re-equalize the pressures on both sides of the check valves when the pressure on the CMT side may be higher than that on the pressurizer side and the check valves prevent the backflow. EPRI indicates that overpressure protection will be required for the CMT and associated piping systems in accordance with Section 3.4.8, which requires overpressure protection for portions of passive systems that can be isolated between closed valves. EPRI further states that the plant designer intends to use CMT discharge valves with pressure under the seat. The valve operators will be sized so that the valve will open if the CMT pressure exceeds RCS pressure by 200 psi, thus providing the required protection. The staff will evaluate this during the review of an individual application for FDA/DC.

The passive Requirements Document does not contain guidelines on design consideration of several phenomena that could affect the performance of the CMT in injecting flow into the reactor vessel, such as possible flow restrictions as a result of condensation in the pressure equalization lines, possible dynamic effects of steam condensation in the CMT and thermal stress of the CMT because of thermal cycling resulting from ingress of high-temperature pressurizer steam and condensation, and thermal stratification in CMT. In response to staff questions, EPRI acknowledged that steam condensation in the equalization line and CMT can affect CMT injection flow and that thermal stress resulting from ingress of high-temperature fluid needs to be accounted for in the mechanical design of the CMT. It further stated that a large-scale test of the CMT and associated piping is planned by the plant designer to understand these phenomena, determine the extent of condensation, and develop models that account for condensation in safety analysis and accommodate them in the design. The staff will review this test program during the review of an individual application for FDA/DC.

5.3 PWR Passive Decay Heat Removal System

The PDHR system will be designed to remove core decay heat whenever the steam generators are not available, during safe shutdown, or if reactor shutdown

cooling is unavailable with the reactor vessel head still in place during normal cooldown. The PDHR system will consist of heat exchangers located within the IRWST at an elevation above the core. It will remove heat by natural circulation of primary coolant from the RCS through the heat exchangers and return cooled water to the RCS. Heat rejected to the IRWST will initially heat up the IRWST water and, once boiling is initiated, will remove the water through the passive containment cooling system (PCCS).

In the long term following a LOCA, the IRWST water will refill the RCS to the level of the break. The IRWST water discharged out of the break will fill the containment building above the RCS loop piping. Steam produced by core decay heat will be released to the containment through the DPVs or the break and will be condensed by the passive containment cooling system, and the condensed water will be returned to the reactor vessel through the IRWST, containment sump, and/or directly through the break. Thus, a close coupling will be established between the RCS, the IRWST, the PDHRS, the PCCS, and the containment. This close coupling requires analysis codes with capabilities to perform both the heat transfer and mass inventory tracking. Even though Section 2.5 of Chapter 1 requires that the analyses for licensing design-basis events be performed using the NRC-approved methods, it is necessary to perform integral tests to both validate the codes used for analysis and demonstrate the effectiveness of the systems. In response to a staff question, EPRI stated that the designer would have to perform the testing needed to validate the codes used for analyses and demonstrate the effectiveness of the systems. It further stated that the Utility Steering Committee was continuing to follow the analyses and testing efforts by the designers. A scaled low-pressure integral test of the long-term core cooling capability was planned, and the test program to be carried out would be the subject of specific meetings and discussion with the NRC staff. This will be resolved as a part of the review of an individual application for FDA/DC.

Section 5.3.2 of Chapter 5 specifies the PDHR system performance requirements. It requires the PDHR system to have sufficient capacity to prevent water relief from pressurizer safety valves during a loss-of-normal-feedwater transient, assuming normal secondary-side heat removal is not available. Section 5.3.2.2 originally required the PDHR system to have sufficient capacity to reduce coolant temperature to 400 °F within 72 hours of reactor shutdown and stated that the RCS at 400 °F was considered an adequate end state for passive plant safe shutdown.

In Revision 4, Section 5.3.2.2 of Chapter 5 was revised to require the PDHR system to have sufficient capacity to reduce coolant temperature to 420 °F within 36 hours of reactor shutdown assuming the most limiting single active component failure. Although not completely complying with the RG 1.139 requirement to bring the reactor to a cold-shutdown condition within 36 hours of reactor shutdown, the requirement in the passive Requirements Document that the PDHR system have the capacity to reduce the reactor coolant temperature to 420 °F within 36 hours of reactor shutdown is consistent with the intent of RG 1.139 for a safety-related RHR system to be capable of bringing the reactor to a safe shutdown condition within a reasonable period of time, assuming the most limiting single failure. The staff evaluates the adequacy of this end state for possible plant safe shutdown in Section 3.3 of this chapter. The staff concludes that 420 °F is an acceptable safe shutdown condition, provided there are acceptable passive safety system performance and appropriate

regulatory oversight measures to ensure availability of risk-significant non-safety systems that are required for maintaining long-term stable shutdown conditions. During its review of an individual application for FDA/DC, the staff will review relevant evaluations to verify the safe shutdown condition.

Section 5.3.3.1 of Chapter 5 specifies the requirements for the PDHR system arrangement. It requires that the PDHR arrangement ensure that the required reactor coolant flow to and from the heat exchanger can be achieved by natural circulation. It requires the PDHR system to contain a single suction connection and single discharge connection from and to the RCS and to have sufficient component redundancy to ensure PDHR system performance even if any automatically operated valve fails to operate and a PDHR heat exchanger is isolated. The single valve failure assumption is a licensing design-basis requirement, and the heat exchanger isolation ensures system operation in the event of a tube leak. Under the emergency core cooling system rule (10 CFR 50.46), a LOCA analysis is required to encompass a postulated break of the PDHR single suction or discharge line that would incapacitate the PDHR function. The PDHR system is not required to function during a LOCA and no credit is taken for LOCA analyses.

In the DSER for Chapter 5, the staff concluded that an arrangement of deep vertical heat exchanger tubes in the IRWST might result in boiling suppression at the bottom of the IRWST as a result of static head, which could result in reduced PDHR heat transfer. EPRI has stated that testing of the PDHR system had included full-height tubes with the full static level of IRWST water, and, accordingly, the heat transfer correlations developed from such testing accounted for the effects of suppressed boiling at the bottom of the IRWST. This testing is the subject of ongoing discussions between the NRC staff and a prospective ALWR vendor. The staff recognizes the ALWR Program policy stated in Section 2.8 of Volume 1 requiring that successful, proven technology be employed throughout the plant, including designs of systems and components. During its detailed review of an individual application for FDA/DC, the staff will evaluate and verify that the actual design properly accounts for the effects of possible boiling suppression and provides for adequate heat transfer for the PDHR system.

5.4 PWR Depressurization System

The depressurization system (DPS) will be a dedicated safety system designed to provide a safety-grade means to depressurize the RCS to permit lower pressure passive safety injection following a design-basis accident and to vent noncondensible gases from the pressurizer. The system will include the valves and piping needed to establish flow paths from the pressurizer steam space to the IRWST and containment atmosphere.

Section 5.4.2 of Chapter 5 specifies the performance requirements for the DPS. It requires the DPS to have sufficient vent capacity to reduce RCS pressure to permit safety injection, assuming the limiting single failure. It requires that means be provided to minimize the likelihood of DPS initiation and that sufficient RCS inventory and/or makeup be provided so that DPS initiation will not be required for 8 hours following an RCS leak, that is, 10 gpm of cold fluid in excess of available charging capacity. The intent of this requirement is to allow the plant operator enough time for recovery from a small leak without actuating the DPS. In a letter dated July 1, 1991, EPRI stated that a

leak of 10 gpm was based on the assumption that it would be permissible for the plant to be operating within the 10-gpm leakage allowed in current standard technical specifications if the charging pumps became unavailable. Eight hours was considered a reasonable period of time for an operator to restore a charging pump to service.

The staff concludes that the passive Requirements Document requirement of 10 gpm in excess of charging capacity is not consistent with the standard technical specifications permissible leakage of 10 gpm during plant operation. The 10-gpm identified leakage limit specified in standard technical specifications can be made up by the charging system and therefore will not result in uncovering of the core, whereas 10 gpm in excess of charging capacity may result in RCS inventory loss and subsequent depressurization. However, the passive Requirements Document design objective (to have sufficient RCS inventory and/or makeup to delay DPS initiation by 8 hours for a leakage of 10 gpm in excess of available charging capacity) does not conflict with Commission regulations because a design-specific small-break LOCA analysis, which encompasses a break size corresponding to the proposed RCS leakage rate and considers the RCS inventory, CMT water inventory and injection, and the delayed DPS actuation, is required to demonstrate acceptable consequences. Therefore, the passive Requirements Document design objective is acceptable; however, the staff will review the results of the LOCA analysis to confirm its compliance with 10 CFR 50.46 during its review of an individual application for FDA/DC.

Section 5.4.2.1.3 of Chapter 5 originally required the PSIS capacity to be sufficient to prevent DPS initiation for 30 minutes following a steam generator tube rupture (SGTR). The staff requested additional information regarding the lack of a requirement to perform necessary analysis to show compliance with radiological dose release criteria when DPS initiation is delayed for 30 minutes. In response to this request, EPRI stated that the designer was required to show compliance with radiological dose release criteria, without credit for any operator action to avoid DPS actuation following an SGTR, as the licensing design-basis analysis of a SGTR is called for in Table 1.2-1 of Chapter 1.

The staff concludes that the requirement in Chapter 1 of the passive Requirements Document for an SGTR licensing design-basis analysis, taking no credit for operator action to demonstrate compliance with the radiological consequence criteria, satisfies the NRC regulation. The requirement in Section 5.4.2.1.3 of Chapter 5 to prevent DPS initiation was intended to be a safety margin basis requirement that the operator have 30 minutes or more to act to prevent plant conditions that could result in a significant plant outage. As this best-estimate evaluation is not a licensing design-basis analysis, appropriate operator actions can be credited for margin assessment.

In Revision 4 of the passive Requirements Document, Section 5.4.2.1.3 of Chapter 5 was revised to specify that such system capability be provided that DPS initiation is avoided following the double-ended rupture of a single steam generator tube. This is to be evaluated on a best-estimate (BE) basis, assuming no operator action after reactor trip. This requirement is consistent with the SGTR coping capability of the plant systems specified in Section 4.2.5.1.3 of Chapter 3, which requires that means be provided to automatically equalize primary and secondary-side pressures following a single

SGTR, without water relief via the secondary-side safety valves of power-operated relief valves, without operator action, and without initiating the automatic depressurization system. As discussed in Section 4.1 of Chapter 3 of this report, the safety margin basis demonstration of this coping capability on a BE basis by taking credit for non-safety systems, such as the control grade turbine bypass system, is acceptable. However, during its review of an individual application for FDA/DC, the staff will require the licensing design-basis analysis of SGTR events that takes credit for only the safety systems without operator action to demonstrate that the radiological consequence is within the acceptable limit.

With regard to DPS redundancy, Section 5.4.3.1.1 of Chapter 5 of the passive Requirements Document requires the DPS to have two valve groups to discharge reactor coolant from the pressurizer to the IRWST and to provide a final stage to discharge reactor coolant from the RCS hot leg to the containment atmosphere. This section originally stated that the final stage would consist of a single discharge line from the hot leg and two parallel lines, each containing two DPVs. The intent was to ensure that a single valve failure would not prevent depressurization. The staff recommended that a separate drop line from the hot leg be used for each group of final-stage DPVs to protect against any potential line blockage that would incapacitate DPV function. EPRI stated that the licensing design-basis analysis in accordance with the NRC requirements had historically not been based on the assumption of passive failure in the short term; therefore, a separate line was not considered necessary. In the DSER for Chapter 5, the staff stated that since depressurization through the final stage of DPVs was important for PSIS injection, it believed that a separate line was necessary. This was identified as an open issue. In Revision 4, EPRI revised Section 5.4.3.1.1 by deleting the requirement for a single discharge line from the hot leg and Figure 5.5-1 has been revised to include parallel drop lines from the hot leg to the parallel DPV sets. This resolves the staff's concern and is acceptable. Therefore, this open issue is closed.

SECY-93-087 gives the staff position on DPS reliability. As part of severe-accident mitigation and to reduce the potential for containment failure as a result of direct containment heating from high-pressure core debris, the Commission paper gives the general criterion that the passive plant design must include a reliable DPS. The passive Requirements Document addresses reliability for the passive PWR DPS by including requirements to provide 100-percent system redundancy (5.4.3.1.1), two depressurization valves in series (5.4.4.1.2), a system design that will minimize the potential for inadvertent initiation during anticipated transients (5.4.2.1), and automatic initiation and sequence of opening for postulated LOCAs of different sizes and locations (5.4.4.2). The 100-percent system redundancy allows the system to perform its function given a single failure, while having two valves in series allows the line to be isolated should a valve fail open minimizing this event's potential as a LOCA initiator. The general requirements for designing to prevent inadvertent initiation and providing automatic initiation and sequencing will help ensure reliable operation of the system during normal operation as well as on demand for the system actuation. In addition, the passive Requirements Document criterion also prohibits providing any means of manually defeating the system, although manual control of the valves will be available. The passive Requirements Document also requires two of four logic to be used for initiation signals to help reduce the potential for inadvertent actuation.

While the above requirements all address system reliability, this section of the passive Requirements Document fails to address testability of the control system and valve actuators for the DPS. It is the staff's position that in order to ensure reliability of the DPS, testability of the control/protection system must be provided in order to provide for regular surveillance testing. Upon further review, the staff concludes that the testability requirement of Chapter 10, Section 3.6 for testing of protection/control circuitry is applicable to the DPS and is acceptable. In addition, the testability requirement of Chapter 1, Section 12.2.7 for testing of valves and valve actuators is also applicable to the DPS and is acceptable. Therefore, the staff concludes that the requirements for the PWR DPS are adequate to meet the staff position in SECY-93-087 regarding DPS reliability.

6 MITIGATION REQUIREMENTS

6.1 Introduction

Section 6 of Chapter 5 of the passive Requirements Document specifies mitigation requirements applicable to both BWRs and PWRs. Mitigation relies on two functions: (1) the containment integrity function and (2) the fission product control function. The containment is intended to provide a barrier to the uncontrolled release of radioactivity in the event of an accident.

6.2 Containment Isolation System Design

The function of the containment isolation system is to permit the normal and emergency passage of fluids through the containment boundary while preserving the capability of the boundary to prevent or limit the escape of fission products that may result from postulated accidents. The containment isolation system will include the portions of all fluid systems penetrating the containment that will perform the isolation function. Section 6.2.2.1.1 of Chapter 5 states that acceptance bases for isolation provisions in lines that penetrate the containment boundary will be in accordance with ANS 56.2, Section 3.6, "Other Defined Bases," and RG 1.141, "Containment Isolation Provisions for Fluid Systems." It was not clear which version of ANS 56.2 was referenced. In its response to a staff request for additional information, EPRI stated that the 1976 version of ANSI/ANS 56.2 was the intended reference and that the passive plants will be in compliance with SRP Section 6.2.4, "Containment Isolation System." The staff has approved ANSI/ANS 56.2-1976. Therefore, EPRI's requirements related to ANS 56.2 are acceptable.

General Design Criteria (GDC) 55 and 56 of Appendix A to 10 CFR Part 50 require that each line that penetrates the containment and is part of the reactor coolant pressure boundary or is connected directly to the containment atmosphere have one isolation valve inside and one isolation valve outside the containment, unless it can be demonstrated that the design is acceptable on some "other defined basis." Each valve must be automatic or locked closed. For satisfying the requirements of GDC 55 and 56, Section 3.6 and Appendices A and B of ANSI/ANS 56.2-1976 provide guidelines that the staff has found acceptable.

The passive Requirements Document specifies that remote manual valves, instead of automatic valves, will be used for essential lines that are not part of engineered safety systems. Justification will be provided for each use of a remote manual valve. The justification will include an evaluation identifying the indications and timing under which isolation must be initiated for inclusion in off-normal operating procedures.

Item II.E.4.2 of NUREG-0737 "Clarification of TMI Action Plan Requirement," and NUREG-0718 "Licensing Requirements for Pending Application for Construction Permits and Manufactory License," require that the systems penetrating the containment be classified as either essential or nonessential. RG 1.141 provides guidance on the classification of system lines. Generally, essential systems are lines in engineered safety feature (ESF) or ESF-related systems needed for safe shutdown of the plant. These essential lines may include remote manual valves, but provisions should be made to detect possible leakage

from these lines outside the containment. Since the use of remote manual valves will be justified on a case-by-case basis to meet regulatory requirements, the staff concludes that EPRI's position is acceptable. The staff will review specific ALWR plant designs referencing the passive Requirements Document against SRP Section 6.2.4.

Section 6.2.2 of Chapter 5 of the passive Requirements Document states that Type C leakage rate testing may be avoided for valves in water-sealed lines that terminate in closed systems outside the containment if the systems are designed to qualify as extensions of the containment in accordance with Section 3.6 of ANSI/ANS 56.2. The staff expressed its concern about eliminating the Type C testing requirements for valves in a closed system outside the containment that meets the criteria of Section 3.6 of ANSI/ANS 56.2-1976 to qualify as a second containment isolation barrier. While a closed system outside the containment can be considered a second containment isolation barrier, each barrier (i.e., the single isolation valve at each penetration and the closed piping system outside the containment) is subject to Type C leak rate testing. In its response to this concern, EPRI indicated that the requirements for water-sealed lines in Section 6.3.2.2 of Chapter 5 of the passive Requirements Document also must be satisfied to avoid Type C testing.

In the DSER for Chapter 5, the staff stated that its position was that for water-sealed valves meeting conditions described in Section 6.3.2.2 of Chapter 5, Type C leakage rate testing with water was acceptable and the test results need not be added to the Type C leakage rate test total. However, this did not justify EPRI's position stated in Section 6.2.2 of Chapter 5 of the passive Requirements Document that Type C testing of these valves could be eliminated. Therefore, the staff concluded that EPRI's position on eliminating Type C testing for the water-sealed lines based on closed systems outside the containment was unacceptable. This was identified as an open issue. In its letter dated April 30, 1993, EPRI revised Section 6.2.2.2 of Chapter 5 by deleting the reference to the elimination of Type C testing for systems that are closed outside the containment. The staff concludes that this revision is acceptable; therefore, this open issue is closed.

In Section 6.2.2.2 of Chapter 5, EPRI states that Type C testing is not required for PWR main steam, feedwater, emergency feedwater, or steam generator blowdown isolation valves. These isolation valves are associated with secondary systems that are closed inside the containment. EPRI states that the closed system inside the containment will preclude the containment atmosphere from reaching the associated isolation valves; therefore, in accordance with GDC 57, only one isolation valve outside the containment is required for each containment penetration. However, in case of steam generator tube leakage, these isolation valves become the only isolation barrier for those containment penetration lines discussed above. Therefore, in the DSER for Chapter 5, the staff stated that its position was that those valves must be leak tested with operating fluid. This was identified as an open issue. The staff will determine whether it is appropriate to require Type C testing for secondary systems that are closed inside the containment during its review of an individual application for FDA/DC.

In Section 6.2.2.3.1 of the passive Requirements Document, EPRI states that isolation valve closure times will be in accordance with ANSI/ANS 56.2-1984

for standard commercial valve operators. This requirement is essentially the same as that of ANSI/ANS 56.2-1976, which has been approved by the staff and, therefore, is acceptable.

6.3 Containment Leakage Rate Testing

Section 6.3.1.2 of Chapter 5 of the passive Requirements Document states that containment leakage rate testing provisions are designed to permit periodic verification that the containment leakage rate is within appropriate acceptance criteria. The passive Requirements Document also states that containment leakage rate testing will be performed in accordance with regulatory requirements, such as Appendix J to 10 CFR Part 50, and the test methods will be in accordance with ANSI/ANS 56.8, "Containment System Leakage Rate Testing Requirements." Furthermore, the passive Requirements Document states that Appendix J requirements will take precedence in the event of any conflict between Appendix J and ANSI/ANS 56.8, except items identified in Section 6.3.2.2 of Chapter 5 of the passive Requirements Document. Any deviations from Appendix J would require issuance of an exemption to the regulation at the time of FDA/DC. The staff's evaluation of these items is given below.

Containment Integrated Leak Rate Test

In Section 6.3.2.2 of Chapter 5, EPRI states that integrated leakage rate tests (ILRTs) can proceed to completion if a leak occurs during testing, provided the leak can be isolated, subsequent repairs are performed, and local before-and-after leakage rate tests demonstrate that the integrated leakage rate acceptance criteria are met. The staff concludes that this position is consistent with practice at current operating plants to meet the requirements of Appendix J.

Type B Air-Lock Tests

In Section 6.3.2.2 of Chapter 5, EPRI states that air locks that are not used during a 6-month period may be tested at containment design pressure after the next usage rather than at 6 months. Air locks opened during periods when containment integrity is not required need to be tested only at the end of these periods.

EPRI states that the air-lock test interval is a plant optimization issue that deviates from the current requirements of Appendix J but is consistent with the proposed rule change. In the DSER for Chapter 5, the staff identified the acceptance of the air-lock test interval as an open issue. The staff proposed final rule changes to Appendix J to 10 CFR Part 50 and transmitted these changes to the Commission in SECY-91-348, "Issuance of Final Revision to Appendix J to 10 CFR Part 50, and Related Final Regulatory Guide 1.XXX (MS 021-5)." This document proposes modification of the regulation that addresses several issues raised by EPRI, including extension of the airlock test interval. This proposed rulemaking was withdrawn and is currently being reexamined under the Commission-approved program "Elimination of Requirements Marginal to Safety." Once the staff has completed this reexamination, it will recommend a course of action to the Commission.

In addition, the staff had previously granted exemptions from Appendix J regarding similar air-lock test intervals for operating plants on a case-by-case basis. The staff has reevaluated the 6-month test requirement and has developed a revised position that meets the objectives of the Appendix J requirements for containment air-lock tests. This revised position still requires the containment air lock to be tested at 6-month intervals at a calculated peak containment DBA pressure (Pa) in accordance with Appendix J, except this interval may be extended up to the next refueling outage if there have been no air-lock openings since the last successful test at Pa. The intent of the Appendix J requirement is to ensure that the integrity of the air-lock door seal is maintained and that no degradation has occurred because of the opening of the air lock between testing intervals at Pa. On the basis of operational experiences at current plants, the staff believes that no significant safety penalty should be caused by this change to the maximum interval between Type C tests. Therefore, it concludes that the EPRI-proposed air-lock test interval is acceptable and this open issue is closed.

Type C Containment Local Leakage Rate Testing

Section 6.3.2.2 of Chapter 5 states that those valves that are in lines designed to be filled with a liquid for at least 30 days after an accident may be leakage rate tested with a liquid. Liquid leakage is not to be converted to equivalent air leakage nor added to the Type C testing total, but is to be reported as liquid leakage. The staff finds this acceptable providing (1) such valves have been demonstrated to have fluid leakage rates that do not exceed their design leakage rates and (2) the installed isolation valve seal-water system fluid inventory is sufficient to ensure the sealing function for at least 30 days at a pressure of 1.1 Pa (calculated peak pressure). The staff concludes that the EPRI-proposed leakage rate testing for the water-sealed valves is acceptable because it includes the above two conditions.

Appendix J to 10 CFR Part 50 requires that Type C tests be performed during each reactor shutdown for refueling, but in no case at intervals greater than 2 years. The passive Requirements Document requires that the maximum interval between Type C tests be 30 months rather than the 24-month interval currently required by Appendix J. This is a plant optimization issue proposed by EPRI. In the DSER for Chapter 5, the staff stated that this issue was consistent with the proposed revision to Appendix J, which would change the Type C test interval from 24 months to 30 months pending the approval of the Commission. EPRI's proposed interval for Type C testing deviates from the requirements of Appendix J. EPRI states that this modification would allow some margin between the nominal 24-month refueling interval and the Type C test interval to ensure the plant shutdown will not be required solely to perform Type C tests. In parallel with this review, the staff has developed proposed changes to Appendix J to 10 CFR Part 50 for all reactors. These changes were transmitted to the Commission in SECY-91-348 to propose modifications of the regulations that would allow the increased test interval and address other issues raised by EPRI. However, this proposed rulemaking has been withdrawn and is currently being reexamined under the Commission-approved program "Elimination of Requirements Marginal to Safety." In addition, the staff has extended the time interval for performing Type C leakage testing at currently operating plants beyond the Appendix J requirements on a case-by-case basis. In its letter dated May 13, 1993, the Advisory Committee on Reactor Safeguards (ACRS) found no significant safety penalty caused by this change to the

maximum interval between Type C tests and agreed with the proposed staff position. Therefore, the staff has recommended in SECY-93-087 that the Commission approve its position that until the rule change proceedings for Appendix J to 10 CFR Part 50 are complete, the maximum interval between Type C tests for both evolutionary and passive plant designs should be 30 months rather than the 24-month maximum interval currently required in Appendix J. Therefore, this open issue is closed.

6.4 Fission Product Leakage Control (FPLC)

Section 6.4.1 of Chapter 5 of the passive Requirements Document states that the Passive ALWR uses passive systems and natural processes for control of fission products in the event of core damage. This applies both to fission products inside the containment as well as fission products which have leaked from containment. The staff has provided its review of EPRI's requirements for a physically-based source terms and dose limits in Sections 2 of this chapter. In Section 6.4, EPRI has specified requirements for the evaluation that plant designers must perform to justify the reduction in fission product release to the environment that is to be credited, as well as features of the plant which are necessary to meet the dose limits.

FPLC for the Passive ALWR is accomplished passively by reliance on holdup of all fission products which the FPLC envelope before release to the environment and as natural processes such as gravitational setting for removal of particulate fission products.

In order to achieve its goal of providing sufficient FPLC to maintain the dose consequences of core damage events below specified limits, EPRI has included requirements for simplicity of design, maintenance, and surveillance.

EPRI requires that plant designer perform a detailed evaluation of the FPLC capability to justify the reduction in fission product release to the environment which is to be credited. The evaluation must include: definition of the envelope to be used for FPLC in the secondary structure and whether the FPLC is to be credited for LDP and PAG or PAG dose only, as well as definition of testing and/or analysis which will establish that the allowable leak rate for the FPLC envelope and bypass pathways are met.

As stated in Section 2.1.7 of this chapter, the staff only accepts the BWR safety envelope for providing potential credit for fission product holdup for decay. In the DSER for Chapter 5, it stated that EPRI's radiation dose consequence criteria of the PAGs for design-basis (DBAs) accidents were not acceptable. The staff concluded that the reference values in 10 CFR Part 100 should be used instead and identified this as an open issue. In its letter dated June 8, 1993, EPRI deleted the use of PAG dose limits for DBA dose consequence assessment from Section 2.2.3 of Chapter 5 of the passive Requirements Document and now requires that regulatory limits under licensing assumptions be met. The staff will review individual applications for FDA/DC against the reference values in 10 CFR Part 100. Therefore, this open issue is closed.

Primary Containment Spray

EPRI does not require an active containment atmosphere cleanup system. Therefore, no containment spray system is provided. EPRI states that fission product control should be accomplished passively by reliance on natural deposition (gravitational settling and plateout) and holdup (decay). GDC 41, 42, and 43 of Appendix A to 10 CFR Part 50 require systems to control fission products to reduce the concentration that may be released to the environment. In evolutionary plant designs, the containment spray system reduces containment pressure and temperature and removes airborne radioactive fission products in the containment atmosphere following a LOCA.

The containment sprays are most effective in reducing such airborne aerosols as cesium iodide, and they are most cost effective compared with the other active fission product removal processes that mitigate radiological consequences (e.g., engineered safety feature filtration). A preliminary draft report "A Simplified Model of Aerosol Removal by Containment Sprays," prepared by Sandia National Laboratories (Sandia) for the NRC, indicates that typical PWR containment sprays are capable of rapidly reducing the concentration of radioactive aerosol in the containment atmosphere. For example, the Sandia report concludes that a typical PWR containment spray can achieve a decontamination factor (DF) of 10E4 in less than 1 hour (median value) or in 7 hours with a 90-percentile confidence level, while a DF of 10E2 (equivalent effectiveness of a typical 4-inch-thick charcoal adsorber for the entire 30-day accident period) can be obtained within 30 minutes (median value) or within 2.5 hours with a 90-percentile confidence level. The spray system DF is defined as the aerosol mass initially in the containment atmosphere divided by the aerosol mass present after spray operation for a given time.

EPRI did not provide a radiological consequence assessment to demonstrate that without containment spray, the ALWR passive plant designs can meet the dose reference values of 10 CFR Part 100 and the control room operator dose limits in GDC 19. This was identified as an open issue in the DSER. In its letter dated June 22, 1993, EPRI submitted a preliminary radiological accident assessment to demonstrate the feasibility of a passive design without containment spray. As discussed in Section 2.1.6 of this chapter, the staff will not require that an active containment spray system be provided in the ALWR design, provided the passive ALWR vendor submits an acceptable radiological consequence assessment that demonstrates that the passive ALWR design without a containment spray system can meet the dose reference values in 10 CFR Part 100 and the control room operator dose limits in GDC 19. Therefore, this open issue is closed. Further details of the staff's position are contained in the draft Commission paper dated February 10, 1994, which is provided in Annex C of Appendix B to Chapter 1 of this report.

6.5 Control Room Habitability

GDC 19 of Appendix A to 10 CFR Part 50 states that (1) a control room should be provided from which actions can be taken to operate the nuclear power plant safely under normal conditions and to maintain it in a safe condition under accident conditions including a loss-of-coolant accident and (2) adequate radiation protection should be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the

body, for the duration of the accident. In current plants, safety-grade, filtered control room HVAC systems with charcoal adsorbers are used to ensure that radiation doses to operators will be maintained within the GDC 19 criteria in the event of an accident.

In SRP Section 6.4, "Control Room Habitability Systems," the staff defined the acceptable operator dose criteria in terms of specific whole-body and critical organ doses (5 rem to the whole body and 20 rem each to the thyroid and skin).

Originally, EPRI proposed the exposure limit for control room operators of 5 rem whole body, 75 rem beta skin dose, and 300 rem thyroid dose. EPRI stated that each operator would be provided with individual breathing apparatus and protective clothing, if required, to meet regulatory limits. The staff determined that EPRI did not adequately justify its requirements for the thyroid and beta skin doses. The staff informed EPRI that the long-term use of breathing apparatus during DBAs had never been allowed. More importantly, the long-term use of breathing apparatus was likely to degrade control room operator performance during and after an accident. This was identified as an open issue in the DSER.

Also, EPRI originally provided requirements to ensure that the control room would be designed to be maintained for a 72-hour period as the primary location from which personnel could safely operate in the event of an accident. In the DSER, the staff stated it position that the required duration for certain accident sequences may be much longer than 72 hours during DBAs. GDC 19 states that "adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions . . . for the duration of the accident," which has typically been assumed to be 30 days. Consequently, the staff concluded that analyses of control room habitability should consider the duration of the accident, which might extend beyond the EPRI-proposed 72-hour period as the design basis.

In Revision 4, EPRI proposed an alternative in which a safety-grade pressurization system could be recharged remotely after 72 hours. Section 6.4.5 of Chapter 5 of the passive Requirements Document requires (1) a passive, safety-grade control room pressurization system that would use bottled air to keep operator doses within the limits of GDC 19 and SRP Section 6.4, Revision 2, for the first 72 hours of the event and (2) safety grade connections for the pressurization system to allow the use of offsite, portable air supplies if needed after 72 hours to minimize operator doses. The staff agrees with the concept of a safety-grade pressurization system and EPRI's commitment to limit the operator doses to those specified in GDC 19 and SRP Section 6.4, Revision 2. The staff will evaluate the feasibility and capability of the proposed pressurization systems on a vendor-specific basis.

The staff has reviewed the EPRI's proposed requirements for a safety-grade pressurization system and has determined that:

- The present licensing of nuclear power plants does not require the licensee to have engineered safety feature (ESF) ventilation systems unless the licensee cannot meet the dose criteria associated with the DBAs or other safety criteria. If the licensee cannot meet these criteria, it must ensure that an ESF system or some other safety-grade system is available to mitigate the consequences of a DBA.

- The use of a pressurization system, such as a bottled air system, may not preclude the need for other safety-grade ventilation equipment in the control room. For example, such safety-grade equipment could be required to maintain cooling to the electrical instruments in the control room.

- At least once each refueling cycle, the licensee must demonstrate the adequacy of such a system to pressurize the control room for a 72-hour period and maintain all the other conditions, including temperature, within the acceptable range for the control room envelope. This requirement is consistent with the present requirements for bottled air systems.

The staff agrees with EPRI's concept of the safety-grade pressurization system and the use of safety-grade connections of the system to allow the use of offsite, portable air supplies after 72 hours to minimize operator doses for the duration of the accident. However, a designer must demonstrate (1) the feasibility and capability of the safety-grade pressurization systems to satisfy GDC 19 criteria regarding control room habitability and (2) the availability and capability of the offsite air supplies.

To meet the GDC 19 requirements for radiation protection, both the passive AP600 and SBWR designs include a safety-related pressurization system to maintain at least 31.1 pascal (1/8-inch water gauge (WG)) positive differential pressure. The applicants for the AP600 and SBWR designs also claim that unfiltered leakage into the control room envelope will be restricted to 1.4E-4 to 2.4E-4 cubic meter per second (0.3 to 0.5 cubic feet per minute), respectively. The vendor-specific reviews will be based on the guidelines of SRP Section 6.4, including experience obtained from operating plants concerning (1) the provisions for maintaining and periodically testing for leaktightness to maintain at least 31.1 pascal WG positive pressure relative to all surrounding areas; (2) the adequacy of the ESF filtration system, if needed; (3) the capability of the postaccident safety-related cooling to maintain a habitable environment for control room operators and to ensure equipment operability; and (4) protection inside the control room pressure boundary against the effects of an accidental release of toxic gases and smoke to conform with the requirements of GDC 19.

Each of the passive ALWR designs includes non-safety ventilation systems for the control room envelope. However, these systems are isolated from the control room envelope when the safety-related pressurization system is actuated on a high-high radiation signal and only remain operational for the areas outside this envelope. Therefore, these systems as designed cannot be credited to enhance the safety-related pressurization system during a postulated DBA.

In summary, the staff positions on control room habitability for passive plants are the following:

- The concept of using a passive, safety-grade control room pressurization system that would use bottled air to keep operator doses within the limits of GDC 19 and SRP Section 6.4, Revision 2, for the first 72 hours of the event and safety-grade connections for the pressurization system to allow the use of offsite, portable air supplies is acceptable if needed after 72 hours to minimize operator doses for the duration of the accident.

- Plant designers must demonstrate the feasibility and capability of a pressurization system and the capability and availability of offsite air supplies to maintain control room habitability for the duration of the accident.

- The regulatory treatment of the portable air supply and the non-safety-grade ventilation system should be in accordance with the staff's position in Section A of SECY-94-084 contained in Annex B of Appendix B to Chapter 1 of this report.

The staff will consider the control room habitability of each vendor's design for acceptance. During its review of an individual application for FDA/DC, it staff will evaluate the design for control room habitability to ensure that the requirements in GDC 19 are met and that personnel and equipment in the control room will have a suitable environment for the duration of the accident. Therefore, the DSER open issue associated with control room habitability is closed.

6.6 Combustible Gas Control

Section 6.5 of Chapter 5 of the passive Requirements Document specifies that (1) the means for hydrogen control must be capable of handling an amount of hydrogen equivalent to that generated from oxidation of 75 percent of the fuel cladding surrounding active fuel and (2) the hydrogen concentration inside the containment will be controlled to ensure that the uniformly distributed concentration does not exceed 13 percent under dry conditions or that the atmosphere is rendered noncombustible. In its June 26, 1990, staff requirements memorandum, the Commission stated that the requirements of 10 CFR 50.34(f)(2)(ix) should remain unchanged for evolutionary advanced reactor designs. Additional information concerning the staff's position regarding acceptable implementation of these requirements for passive advanced reactor designs is contained in Section 2.5.3 of Appendix B to Chapter 1 of this report. Although this section of the regulations was originally written for a select group of plants whose construction permits were pending as of February 1982, 10 CFR Part 52 established these requirements to be a minimum standard for future plants. By letter dated September 15, 1988, the ALWR Utility Steering Committee provided further justification for this approach. The staff's evaluation of these issues follows.

The requirements of 10 CFR 50.34(f) specify that a hydrogen control system that can safely accommodate hydrogen generated by the equivalent of a 100-percent fuel-cladding metal-water reaction must be provided in the design of nuclear plants for which the regulation is applicable. Additionally, the regulation specifies that the hydrogen control system must be designed to ensure that uniformly distributed hydrogen concentrations in the containment do not exceed 10 percent or that the postaccident atmosphere will not support hydrogen combustion.

Aside from the issue of regulatory compliance and applicability, the staff concludes that compliance with 10 CFR 50.34(f) remains appropriate for combustible gas control design in ALWRs. Research (discussed in NUREG/CR-4551 "Evaluation of Severe Accident Risk") indicates that in-vessel hydrogen generation associated with core-damage accidents may range from 40 to 95 percent active cladding oxidation equivalent. The amount of cladding oxidation

is dependent on a variety of parameters related to sequence progression, such as RCS pressure, reflood timing and flow rates, as well as core-melt progression phenomena. Thus, a 75-percent-equivalent cladding reaction continues to be viewed as a reasonable design basis for hydrogen generation for severe accidents in which the reactor pressure vessel (RPV) remains intact. However, it is the staff's view that ALWR passive plants should provide protection for hydrogen generation resulting from a wider spectrum of accidents, that is, full core-melt accidents with RPV failure. In that context, it also is necessary to consider ex-vessel hydrogen generation as a result of core debris reacting with available water or core-concrete interactions. Calculations using the CORCON models indicate that if the core debris is cooled in relatively rapid fashion (1 to 2 hours), additional hydrogen generation will be less than that equivalent to a 25-percent cladding oxidation reaction. This relatively limited ex-vessel reaction is conditional on the existence of a coolable debris bed and the availability of sufficient water. If extensive core-concrete interaction occurs as a result of no cavity flooding, more hydrogen generation should be considered. Considering the effects discussed above, the staff concludes that an equivalent 100-percent cladding oxidation reaction is an appropriate deterministic design criterion and a reasonable surrogate for the combination of both in-vessel and ex-vessel hydrogen generation.

EPRI submitted a containment failure prevention study to provide additional information in support of the passive Requirements Document. That study provided some further discussion on the basis for EPRI's position on hydrogen within the containment during a severe accident. The study indicated that the 75-percent metal-water reaction and 13-percent concentration will be one of the criteria for sizing the containment; however, to meet 10 CFR 50.34(f), EPRI would develop a new licensing design-basis document. Although the study indicated that EPRI planned to meet 10 CFR 50.34(f), this position was not clearly stated. The staff's position is that if the licensing design basis required analysis to demonstrate that the containment performance would meet 10 CFR 50.34(f) with a 100-percent reaction, maintain a 10-percent concentration, and consider any effects resulting from hydrogen control system operation, this would be consistent with the NRC position on combustible gas control.

In Revisions 4 and 5 to the passive Requirements Document, Sections 2.4 and 6.5 of Chapter 5 were modified as follows:

(1) Section 2.4.1.7 states: "The design shall ensure that containment integrity is maintained during an accident that releases hydrogen generated from 100% metal-water reaction of the fuel cladding accompanied by hydrogen burning or the added pressure from post-accident inerting."

(2) System Section 6.5.3.1 requires that for passive PWRs, a non-safety-grade passive autocatalytic recombiner (PAR) system will be provided that will "ensure margin in the design for natural circulation mixing of the containment atmosphere by ensuring that local hydrogen concentrations, except at the location of release, will not exceed 10 percent" and "the PAR system shall utilize devices with demonstrated capability to combine hydrogen and oxygen under severe accident conditions at a sufficient rate and in a manner compatible with the passive ALWR concept."

(3) Section 6.5.4.1.2 requires that passive BWRs rely on an inerting system to meet the combustible gas control requirements stated above. "The inerting system shall be capable of reducing the wetwell and drywell bulk oxygen concentrations from atmosphere conditions 3.5 volume percent in less than four hours and maintaining that concentration or less during normal plant operation." This requirement is designed to provide "margin to ignitable hydrogen-oxygen ratios for the maximum range of design basis hydrogen generation." In addition, under Passive Autocatalytic Recombiner System Performance Requirements, Section 6.5.3.1.2: "A non-safety grade PAR system shall be provided for a BWR inerted containment to prevent deinerting of either the drywell or the wetwell from net oxygen production during a severe accident without reliance on any other combustible gas control device."

These new requirements satisfy the requirements of 10 CFR 50.34(f)(2)(ix) which the staff has determined are applicable to the passive ALWRs. Both plant designs are required to accommodate hydrogen generation equivalent to a 100-percent metal-water reactor of the fuel cladding. The passive PWR design will provide a hydrogen control system that will limit hydrogen concentration to less than 10 percent and provide containment-wide hydrogen control for severe accidents. The passive BWR design will meet the requirement by ensuring that the postaccident atmosphere in the containment will not support hydrogen combustion. On this basis, the staff finds the proposed hydrogen control requirements acceptable. Section 6.5.5.2 of Chapter 5 of the passive Requirements Document requires that PAR devices word for DBA mitigation be safety related and Section 6.5.3.1.1 requires that PAR devices used for accident mitigation be non-safety related. In addition, the staff is reviewing the report submitted by EPRI by letter dated April 8, 1993, titled "Qualification of PARs for Combustible Gas Control in ALWR Containments." The staff will evaluate the acceptability of the PAR system for combustible gas control during its review of an individual application for FDA/DC.

The staff notes, however, that PARs are not the only way to acceptably meet the hydrogen control requirements. It still considers igniters to be acceptable. Considering that the staff is still reviewing PARs to determine if they are acceptable for meeting the regulations, it would not be prudent to exclude other acceptable means of hydrogen control. Nor would it be prudent to ever exclude alternative means of meeting regulatory requirements. Therefore, the staff has not deleted from this report previous references to igniter systems and their acceptability. Nor should it be construed that PARs are the only acceptable means of hydrogen control for any of the advanced designs.

If a deliberate ignition system (i.e., igniters) is selected for combustible gas control, safety-related equipment will be required to be capable of surviving the potential deflagrations to which it might be exposed. The passive Requirements Document does not provide guidance regarding the timing of igniter activation in the event of an accident. It is the staff's position that such a deliberate ignition system should be activated early in the accident sequence, but no later than before local or global detonable concentrations develop (either directly or through deinertion of the containment by containment spray and/or steam).

As stated above, the passive Requirements Document provides the option of using an inert atmosphere as a means of combustible gas control. If the plant

designer chooses an inert atmosphere in the containment, the designer must ascertain if during postaccident conditions the amount of oxygen generated by radiolysis or introduced from other sources will not produce oxygen concentrations that would deinert the containment atmosphere so that deflagration or even detonation of the accumulated hydrogen could take place. The amount of oxygen generated from these sources depends on plant design, and plant-specific analysis will be required during the facility design phase. The staff will evaluate the plant-specific analyses during its review of an individual application for FDA/DC to ensure compliance with this position. At that time, the system to maintain an inert atmosphere in the containment also will be evaluated.

The system must be designed to maintain the inert atmosphere within the containment for 30 days following the postulated design-basis accident. For severe accidents, the system must be capable of maintaining the inert atmosphere within the containment as long as containment integrity is maintained following a postulated severe accident. The system must be designed to meet the design criteria of GDC 41, 42, and 43 in Appendix A to 10 CFR Part 50.

Combustible Gas Control for Design-Basis Accidents

EPRI has committed in Appendix B to Chapter 1 of the passive Requirements Document to comply with the staff review guidance in SRP Section 6.2.5, "Combustible Gas Control in Containment"; 10 CFR 50.44; 10 CFR 50.46; and GDC 5, 41, 42, and 43. In addition, EPRI considers RG 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," to be an optimization subject as related to source term.

Section 6.5.2.1 of Chapter 5 originally stated that a means of combustible gas control would be provided for the licensing design basis as specified in Section 2.4.2.6 of Chapter 5. In the referenced section, EPRI stated that for the PWR dry containment the uniformly distributed hydrogen gas concentration in the containment did not exceed 13 percent under dry conditions. In the DSER for Chapter 5, the staff concluded that the hydrogen concentration of 13 percent for the PWR dry containment was not consistent with SRP Section 6.2.5 or the limit specified in Table 1 of RG 1.7, which specifies a hydrogen concentration of 4-percent flammability limit for a design-basis LOCA. The staff concluded that EPRI should provide justifications for this deviation and this was identified as an open issue.

In Revision 4 to the Requirements Document, EPRI revised Section 2.4.1.3 of Chapter 1 and Section 2.4.2.6 of Chapter 5, to include a licensing basis requirement that a combustible gas control system be provided that will keep combustible gas concentrations in the containment following a core damaging (design-basis) accident below the level at which a deflagration or a detonation could threaten containment integrity. The rationole for these requirements states that the requirements in 10 CFR 50.34(f). These new requirements are consistent with the staff's position in SECY-93-087, contained in Annex A of Appendix B to Chapter 1 of this report, and with 10 CFR 50.34. Therefore, the staff concludes that they are acceptable and this open issue is closed.

The design of passive plants relies on convective mixing and the physical arrangement of compartments and components (including PARs) inside the containment in order to meet the requirements of 10 CFR 50.44 and 50.46, to

provide the capability for ensuring a mixed atmosphere in the containment, and of GDC 41, to provide systems as necessary to ensure that containment integrity is maintained. Therefore, plant designers must provide a plant-specific analysis to demonstrate adequacy of the mixing in an individual application for FDA/DC. The analysis method and acceptance criteria should meet the analysis requirements in Section 2.5.1 of Chapter 1 of the passive Requirements Document.

Section 2.4.2.6 of Chapter 5 states that for the BWR, an acceptable alternative approach is to provide a noncombustible containment atmosphere (an inert atmosphere). Section 6.5.5.2 of Chapter 5 states that for the BWR, the plant designer will ascertain, on the basis of plant-specific analyses, if during postaccident conditions the amount of oxygen generated by radiolysis or introduced from other sources will not produce oxygen concentrations that will deinert the containment atmosphere. PARs used for DBA mitigation must be safety grade. Additionally, for the long term, the plant designer will make provisions for reducing and disposing of residual hydrogen as part of accident recovery. The staff concludes that EPRI requirements for BWR combustible gas control do not conflict with current Commission regulations.

Section 6.5.4.4 of Chapter 5 originally provided guidance to the plant designer on the design of recombiners that are not safety grade. This was in contrast to the current generation of LWRs and the evolutionary ALWRs in which the recombiner system is safety grade. EPRI's guidance for the recombiner system included non-safety-grade portable recombiners, dedicated containment connections and space for two recombiners, capability to install the recombiners within a specified time, suitable shielding and proper access, and operation and control of the recombiner from the main control room. In the DSER for Chapter 5, the staff stated that it might not require that the hydrogen recombiner system meet all the safety-grade criteria, but there should be a high level of confidence that these safety-important systems are available when needed. This was identified as an open issue.

In its letter dated May 3, 1993, EPRI replaced the non-safety-grade portable hydrogen recombiners with passive autocatalytic hydrogen recombiners (PARs) and revised Section 6.5 of Chapter 5 of the passive Requirements Document. Specifically, Section 6.5.4.4, which provided design provisions for the non-safety-grade portable recombiners, was deleted. Section 6.5.2.3 was revised to require that, for containment atmospheres that are not inerted, capability to mix the atmosphere will be provided by natural circulation so that local detonation of hydrogen can be shown as unlikely to occur. The use of PAR modules may be considered as one means of promoting mixing while also reducing hydrogen concentration. Section 6.5.1.1 was also revised to provide the functional requirements for combustible gas control of the ALWR plant. Combustible gas control includes passive features (i.e., containment volume, atmospheric mixing characteristics, etc.) as well as specific hardware systems (i.e., catalytic modules, inert gas supply, instrumentation and controls). In its letter dated April 8, 1993, EPRI submitted a white paper on PARs titled "Qualification of Passive Autocatalytic Recombiners for Combustible Gas Control in ALWR Containments" to justify their use for combustible gas control.

Current NRC regulations require that containments be designed for the control of hydrogen accumulation following an accident. All applicants for design

certification are required by 10 CFR 52.47(a)(ii) to demonstrate compliance with any technically relevant portions of the TMI-related requirements. The staff is currently reviewing EPRI's submittals on hydrogen control for passive ALWRs and has performed numerous studies and conducted several experimental programs to better understand the behavior of hydrogen combustion and potential concentration gradients within the primary containment. The staff's position is that passive plants should be designed, at a minimum, to the same requirements or those applied to evolutionary designs. The staff will evaluate the ability of passive plant designers using PARs to meet the requirements of 10 CFR 52.47(a)(ii) during its review of an individual application for FDA/DC. Therefore, this open issue is closed.

In its letter dated May 26, 1993, EPRI proposed an approach to determine which systems should receive what kind of regulatory oversight. A key element of this process is to ensure that the passive plant vendors use probabilistic risk assessment methods to examine reliability of the safety and non-safety systems and ensure that important risk insights are considered in the design. This process and the staff's evaluation of it are described in greater detail in Section A of SECY-94-084, contained in Annex B of Appendix B to Chapter 1 of this report. The staff will review implementation of EPRI's process during its review of an individual application for FDA/DC to ensure that the passive plant vendors follow this process satisfactorily.

To ensure containment structural integrity, the staff requested that EPRI consider load combinations associated with a LOCA (i.e., LOCA plus hydrogen burn and safe shutdown earthquake (SSE) plus LOCA) in the passive Requirements Document. In its response, EPRI stated that LOCA plus hydrogen burn and SSE plus LOCA were included in Section 6.6.2.2 of Chapter 5 and Section 4.6.1.1 of Chapter 1, respectively. The staff considers these responses acceptable.

In its response to a staff request for additional information concerning details of the containment design, EPRI stated that Section 6.6.2.2 of Chapter 5 provides stress and buckling criteria for the containment under an accident loading condition involving LOCA and hydrogen burns as described in Section 2.4.2.7 of Chapter 5. For buckling, EPRI proposes that two times the basic compressive allowable stress given in ASME Code Section III, Subsection NE-3222.1, be used. In the DSER for Chapter 5, the staff stated that this limit was not consistent with ASME Code, Section III, Subsection NE-3222.2, which requires 1.2 times the basic compressive allowable limit for the Level C service limit. This was identified as an open issue.

In its letter dated March 31, 1993, EPRI stated that for regions of ellipsoidal or torispherical shell surfaces of the containment, the allowable compressive stress due to internal pressure will be determined in accordance with ASME Code, Section III for Service Level C, supplemented by Code Case N-284. As stated therein, the factor of safety corresponding to 1.67 was proposed. The rationale stated, "Compressive stress in ellipsoidal or torispherical shell heads due to internal pressure loading is a localized stress field which does not represent a challenge to overall containment stability; thus, a lower factor of safety against buckling than otherwise permitted by Subparagraph 3222.2 is appropriate in these regions."

Because of their geometric configurations, torispherical and ellipsoidal shells under internal pressure have a stress field in which membrane tension in the meridian direction and compression in the hoop direction exist with the potential for buckling if not properly designed.

To determine the viability of the containment under a severe-accident condition, it becomes necessary to know the ultimate capacity of the containment more precisely. Buckling is generally perceived to be a problem mainly for thin shells under external pressure as evidenced by the requirements in ASME Code, Section III, Subsection NE. Failure due to buckling under external pressure leads to catastrophic collapse of the shell. However, buckling of the torispherical or ellipsoidal shell heads due to internal pressure can be considered as local buckling. The buckling of such shell heads under internal pressure is of the stable kind. After the first one or two buckles have formed, it is possible to keep on increasing the internal pressure with additional buckles appearing periodically, but there is no effect on the stability of the overall shell. It should be noted that the formation of these circumferential buckling waves on the shell can fracture the joints with any components appended to this portion of the shell and damage the components such as bellows, closure of openings, gaskets, and other attachments. There is also the possibility that the shell wall itself might fracture in the process of forming the buckles, if the shell steel material is brittle. With the continuous increase in the internal pressure and after the cessation of the formation of buckles without any fracture, the shell will most likely fail by axisymmetric yielding.

ASME Code Case N-284 may be applied to evaluate local buckling of stiffened and unstiffened shells under external or internal pressure, stringer buckling, and general instability of stiffened shell under external pressure. The basic comprehensive allowable stress values referred to in NE-3222.1 correspond to a factor of 2 in Code Case N-284. The respective factors of safety for stringer buckling and general stability failures are required to be 20 percent higher than those for critical local buckling. In addition to the factors of safety, capacity reduction factors that account for the effects of imperfections and nonlinearity in geometry and boundary conditions and plasticity reduction factors that account for nonlinearity in material properties are to be applied in accordance with the guidance in Code Case N-280. Further, it should be mentioned that Code Case N-284 has been endorsed in RG 1.84, "Design and Fabriation Code Case Acceptability," Revision 27, with a condition that the effect of the presence of a large opening on the shell should be considered.

Code Case N-284 can be considered as a supplement to NE-3133 and NE-3222 and takes into consideration the local buckling of the shell, whether stiffened or unstiffened, of the stringer buckling and general stability of the stiffened shell as a whole. Such consideration is lacking in either NE-3133 or NE-3222. The factors of safety for local buckling, whether stiffened or unstiffened, for stringer buckling and for general stability failure of stiffened shell are smaller than those for the general buckling of unstiffened shells. This is due to the fact that with stiffeners the shell is less sensitive to imperfections and the stiffened shell has a higher resistance against buckling. Furthermore, local buckling of the shell, whether stiffened or unstiffened, has no effect on the stability of the shell as a whole. Therefore, it would be unnecessarily conservative to use, for local buckling evaluation, the factors of safety specified in NE-3222, which are intended for general

stability of the shell. Code Case N-284 states that the basic factor of safety of 2 is applied to buckling stress values that are determined by classic (linear) analysis that has been reduced by capacity reduction factors determined from lower bound values of test data. It should be noted when Code Case N-284 is applied to shell under internal pressure, the influence of the internal pressure may reduce the initial imperfections.

On the basis of its review and evaluation of Subarticles NE-3100 and NE-3200 and Code Case N-284 on buckling, the staff concludes that the buckling of ellipsoidal and torispherical shell heads due to internal pressure may be considered as local buckling and evaluated on the basis of the criteria contained in Code Case N-284. Therefore, the EPRI's proposed factor of safety of 1.67 for the Level C service limit is acceptable and this open issue is closed. The staff will evaluate the vulnerability of a specific containment design and layout to local buckling because of damage to seals for containments, bellows connections, fracture of shell materials, etc., during its review of an individual application for FDA/DC. Therefore, the open issue concerning stress and buckling criteria for the containment is closed.

6.7 Severe-Accident Requirements

On December 13, 1988, the staff sponsored a meeting with representatives of the nuclear power industry and the general public to discuss the staff's and industry's approaches to resolving severe-accident issues for ALWRs. During this meeting, the staff presented alternative approaches that were being considered to address various severe-accident challenges. Since the December 1988 meeting, the staff has continued to develop its positions on severe-accident criteria. The ALWR Utility Steering Committee's approach to resolving these issues as described in Section 6.6 of Chapter 5 of the passive Requirements Document has been evaluated for consistency with the current staff positions. The staff's evaluation of those features, applicable to both PWRs and BWRs, to address severe-accident concerns follows.

Containment Performance

For severe accidents, the staff's position on containment performance is discussed as Issue I.J of SECY-93-087, provided in Annex A of Appendix B to Chapter 1 of this report. As stated therein, the containment should maintain its role as a reliable leaktight barrier by ensuring that containment stresses do not exceed ASME Service Level C limits for metal containments, or Factored Load Category for concrete containments, for approximately 24 hours following the onset of core damage under the more likely severe-accident challenges, and that following this period, the containment should continue to provide a barrier against the uncontrolled release of fission products. Section 2.6 of Chapter 1 of the passive Requirements Document requires that the "ASME limits specified in Chapter 5, Section 6.6.2.2 should not be exceeded for a period of approximately 24 hours after the start of release of fission products from the fuel." In addition, it requires that "beyond approximately 24 hours means for preventing uncontrolled fission product release from containment shall be provided in accordance with Chapter 5, Section 6.6.2.5." That section requires long-term containment overpressurization protection in support of preventing an uncontrolled release of fission products. This section also indicates that long-term overpressurization protection can be provided by providing a large strong containment (e.g., the designer demonstrating that

the ASME limits of Chapter 5, Section 6.6.2.2, are not exceeded for approximately 2 to 3 days after the accident's beginning for the most likely accident sequences). The assumption is that 2 to 3 days provides adequate time for the plant staff to bring the accident under control. The requirements of Chapter 1, Section 2.6, mirror the staff's position in SECY-93-087 and are acceptable. The staff concurs that 2 to 3 days should be sufficient time for the plant staff to bring the accident under control. It is reasonable to assume that if the containment is strong enough to withstand the highest temperature and pressure it might experience during an accident (e.g., performance limits are not exceeded for this timeframe) that long-term protection against an uncontrolled fission product release will be provided. Therefore, the staff finds this acceptable.

A collapse load analysis should be used for evaluating the containment capacity following the initial 24-hour period after the onset of core damage. SRP Sections 3.8.1, "Concrete Containment," and 3.8.2, "Steel Containment," specify that an analysis be performed to determine the ultimate capacity of the containment, and that a report be submitted documenting the analysis, including the failure mode and the criteria used to establish failure. For a steel containment, Section 6.6.2.4 of Chapter 5 of the passive Requirements Document defines the ultimate capacity as the pressure and temperature loading that corresponds to the collapse load defined by the method detailed in Paragraph II-1430 of the ASME Code, Section III, Appendix II. This ASME Code paragraph describes the criterion (or procedure) for determining collapse load in an experimental stress analysis. It is not clear how this criterion for determining test collapse load will be used in an analysis. Plant designers intending to use this criterion should submit to the NRC staff their methods for applying the test collapse load to the final design analysis.

In addition to requirements for steel containments, Section 6.6.2.4 of Chapter 5 defines the ultimate structural capacity of concrete containments as the pressure and temperature loading that produces liner plate strains equal to the liner strain limits of ASME Code, Section III, Subarticle CC-3720 for the factored load category. Section 6.6.2.4 of Chapter 5 requires that this ultimate capacity analysis consider the penetrations and their interaction with the containment, the shield building, and other structures internal or external to the containment that might cause localized failure before the limit load for the overall pressure boundary. The staff concludes that this criterion for concrete containment is acceptable because it is consistent with ASME Code requirements. However, it will require that plant designers submit to the NRC staff a report on how the results from the testing of prototype details or models of prototype details may be used to augment such analysis as stated in Section 6.6.2.4 of Chapter 5.

The staff requested that EPRI consider providing guidance regarding the allowance for corrosion of carbon steel containment boundaries. In its response, EPRI stated that specific requirements concerning allowance for corrosion of carbon steel containment structures were provided in Sections 4.3.4.1.1 and 4.3.4.1.2 of Chapter 6. This issue is addressed in Chapter 6, Section 2.1.1, of this report.

Core-Concrete Interaction/Core Debris Coolability

Commission paper SECY-93-087 gives the staff's position on core-concrete interaction/core debris coolability for the passive ALWR designs. The staff's position is that the passive ALWR designs must meet the following criteria:

(1) Provide reactor cavity floor space to enhance debris spreading.

(2) Provide a means to flood the reactor cavity to assist in the cooling process.

(3) Protect the containment liner and other structural members with concrete, if necessary.

(4) Ensure that the best-estimate (BE) environmental conditions (pressure and temperature) resulting from core-concrete interactions (CCI's) do not exceed Service Level C for steel containments or Factored Load Category for concrete containments, for approximately 24 hours. Ensure that the containment capability has margin to accommodate uncertainties in the environmental conditions from CCIs.

Regarding Criterion 1 above, EPRI has proposed a floor sizing criterion of 0.02 m^2/MWt (Chapter 5, Section 6.6.3.2.1). The staff neither supports nor disputes this criterion and will evaluate the individual applications for FDA/DC to ensure that adequate reactor cavity floor space has been provided. Criterion 2 is addressed in Section 6.6.3.3 of Chapter 5, which is acceptable because it agrees with the position above. Criterion 3 is addressed in Section 4.3.2.6.2 of Chapter 6, which requires the reactor cavity/lower drywell to be designed to avoid direct contact of core debris with the containment boundary that would cause containment failure. The rationale states that on the basis of the most current information available, this requirement would mean providing a 3-foot concrete cover on the containment leaktight boundary. The staff concurs with this aspect; however, containment and equipment structural members should also be protected with concrete unless analysis shows that this measure is not necessary. Therefore, the staff will evaluate the FDA/DC submittals to see if the need for this criterion has been adequately evaluated and, if needed, included in the design. Criterion 4 is addressed in Sections 2.4.3.1 (top-level requirements for accident mitigation) and 6.6.2.2 (containment performance structural evaluation requirements) of Chapter 5 and in Section 2.6.1 (overall requirements for containment performance criteria) of Chapter 1. Section 2.4.3.1 of Chapter 5 requires that the containment structure be designed with enough margin to provide severe-accident protection and that BE analyses be used to ensure adequate margin in the containment design. The staff notes, however, that to do this, the designer will need to ensure that sufficient margin is included in the BE analyses to account for the uncertainties related to CCI. Section 6.6.2.2 of Chapter 5 requires the containment to be designed to meet Service Level C (steel containments) or Factored Load Category limits (concrete containments) of ASME Code, Section III. Section 2.6.1 of Chapter 1 requires that the containment performance limits of Chapter 5, Section 6.6.2.2, not be exceeded for approximately 24 hours after the start of a release of fission products from the fuel. The staff concludes that the passive Requirements Document criteria are adequate to ensure that the vendor meets the staff's position that BE environmental conditions in the containment following an accident

should not exceed the containment performance criteria for approximately 24 hours are, therefore, acceptable. The staff will, however, review individual applications for FDA/DC to ensure that CCI has been adequately addressed in the designers BE analyses.

Cavity/Pedestal-Drywell Configuration

SECY-93-087 gives the staff position on cavity design features for the mitigation of a high-pressure core ejection accident. The passive Requirements Document addresses the criterion for providing cavity design features to decrease the amount of ejected core that could reach the upper containment generically for the passive plants. Section 6.6.3.1 of Chapter 5 requires that an evaluation be performed to confirm that the quantities of ejected core debris that reach the upper containment in the event of a core ejection accident will not be sufficient to jeopardize containment integrity. The problem with the evaluation specified is that it only addresses low-pressure core ejection unless the probabilistic risk assessment identifies higher pressure ejections as risk significant. This does not meet the intention of the staff position, that is, to provide design features for mitigation even though it is a low probability event. The staff does not dispute that this is a low probability event. During the design stage, however, relatively low-cost design features such as ledges or walls that could deflect debris could easily be incorporated into the design without any significant impact on cost. These features could be beneficial in protecting the containment integrity should a high-pressure core ejection accident occur. Therefore, the staff will review individual application for FDA/DC to ensure that the design incorporates features to limit the amount of core debris that can reach the upper containment following a high-pressure core ejection accident.

Severe-Accident Equipment Requirements

Transmitters and other instrument sensors required for severe accidents will be located outside the containment or will be able to operate in the severe-accident environment. In Table B.1-1 of Appendix B to Chapter 1 of the passive Requirements Document, EPRI has committed to compliance with 10 CFR 50.34(f)(2)(xvii). This addresses TMI Action Plan Item II.F.1 (NUREG-0737), which requires that the high-range area radiation monitors located in the containment be accessible for calibration and designed to function under severe-accident conditions. Furthermore, equipment useful for severe-accident mitigation will be designed to perform its identified function during severe-accident conditions and when needed.

Commission paper SECY-93-087 describes the staff's position on equipment survivability during a severe accident and equipment used to mitigate a severe accident. This equipment is not required to be environmentally qualified in accordance with 10 CFR 50.49 or to necessarily meet the requirements in 10 CFR Part 50, Appendices A and B. This position is consistent with the requirements of Section 6.6.5.4 of Chapter 5. However, the staff notes that guidance such as that in Appendices A and B of RG 1.155, "Station Blackout," is appropriate for equipment used to mitigate the consequences of a severe accident. In addition, the systems used to mitigate the severe-accident must survive in the severe-accident environment and be provided with an alternative power supply (in addition to the normal Class 1E power supply). Because the proposed passive designs do not use Class 1E power supplies, the plant

designer must demonstrate that the passive design power supply for severe-accident equipment is reliable. During its review of the credible severe-accident scenarios included as part of an individual application for FDA/DC, the staff will evaluate the ALWR vendor's identification of equipment needed to perform mitigative functions and the conditions under which the mitigative systems must operate. Equipment identified as having a mitigative function and requiring motive power will be reviewed to determine if there is a high degree of assurance that the motive power will be available and that the equipment will function as required when needed for as long as it is needed. Therefore, the designer must define the environment in which mitigative equipment will be expected to operate and provide the staff with an analysis demonstrating that severe-accident mitigative equipment will function when needed with a high degree of assurance.

7 BWR MITIGATION/CONTAINMENT REQUIREMENTS

7.1 Introduction

Section 7 of Chapter 5 of the passive Requirements Document specifies mitigation requirements applicable to the BWR version of the ALWR. It states that the containment system for the ALWR BWR will include a pressure-suppression pool, drywell, wetwell airspace, and drywell/wetwell vent system in a steel-lined, reinforced-concrete containment structure. Passive containment cooling capability will be provided by the passive decay heat removal (PDHR) system and will be capable of removing heat from the primary containment atmosphere to maintain containment pressure within design limits in the event of a licensing design-basis event. The staff will review the adequacy of containment heat removal capability when detailed design analysis data are available from plant designers.

A reactor building is integral with and surrounds the primary containment, serving as a fission product leakage control barrier. The containment system, operating in conjunction with other plant systems, must limit fission product leakage from a postulated loss-of-coolant accident to values no greater than those required to meet both the control room operator limits of GDC 19 and the offsite dose limits of 10 CFR Part 100.

7.2 Performance Requirements

The BWR secondary containment function will be provided by the fission product leakage control system described in Section 6.4 of Chapter 5 and Section 4.4 of Chapter 6 of the passive Requirements Document. Section 7.2.1 of Chapter 5 states that the containment design conditions will be based on the limiting double-ended guillotine break. It states that leak-before-break methodology may be used, where possible, in subcompartment pressurization analyses consistent with the "broad scope rule" modification of GDC 4 (Ref: 53 FR 11311, Supplementary Information). However, it should be noted that the staff does not accept leak-before-break methodology for carbon steel piping.

Suppression Pool and Vent System

Sections 7.2 8, 7.2.11, and 7.2.16 of Chapter 5 state that the design of the vent system and the suppression pool will be based on proven technology and existing test data and analytical models so that there is no need for additional test models. The staff finds this is acceptable provided the plant designer is able to demonstrate the applicability of the existing test data and analytical models to its design.

Section 7.3.5 of Chapter 5 states that the design of the suppression pool temperature monitoring system will meet the general design requirements specified in NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments." The staff concludes that this commitment is acceptable.

Steam Bypass of the Suppression Pool

The concept of the pressure-suppression containment is that steam released from the primary system will be condensed by the suppression pool, thereby limiting pressurization of the containment system. This will be accomplished by channeling the steam into the suppression pool through a vent system. Bypass leakage paths could exist between the drywell and wetwell airspace that might overpressurize the containment. Potential sources of steam bypass include leakage through the vacuum relief valve, cracking of the drywell concrete structure, and penetrations through the drywell structure. In addition to containment pressurization, the pressure-suppression pool acts as a fission product cleanup system. The bypassed fission product releases carry higher radioactivity, which affects the radiation dose consequences more seriously. Appendix A to SRP Section 6.2.1.1.C, "Pressure-Suppression Type BWR Containments," provides guidance on steam bypass capability for Mark I, II, and III containments.

Section 7.2.26 of Chapter 5 of the passive Requirements Document originally stated that the suppression pool bypass leakage during blowdown through the suppression pool following a design-basis accident would be negligible. In a response to a staff request for additional information, EPRI stated that leakage other than that from the vacuum breakers had been virtually eliminated by an all-welded steel barrier between the wetwell and drywell. A very tight vacuum breaker closure was ensured; therefore, there would be negligible bypass leakage in the ALWR. However, no requirement on the steam bypass capability is proposed in the passive Requirements Document.

In Revision 4 of the passive Requirements Documents, EPRI added a performance requirement that the suppression pool bypass leakage during blowdown through the suppression pool following a design-basis accident be less that the containment bypass capability. Unfortunately, this does not satisfy the staff's position that the containment should have a steam bypass capability consistent with SRP Section 6.2.1.1.C.

According to SRP Section 6.2.1.1.C, the containment should have a steam bypass capability. The allowable bypass leakage is defined as the amount of steam that could bypass the suppression pool without exceeding the wetwell design pressure. The allowable value has to be evaluated by the plant designer for the complete spectrum of credible primary system pipe ruptures. It is expressed in terms of the parameter $(A\sqrt{K})$, where A = flow area of leakage path (ft^2) and K = geometric and friction loss coefficient. This parameter $(A\sqrt{K})$ is dependent only on the geometry of the drywell leakage paths and is a convenient numerical definition of the overall drywell leakage capability.

In the DSER for Chapter 5, the staff concluded that even though the design required negligible bypass leakage, a possibility still existed that vacuum breakers could be left open and could result in steam bypass of the suppression pool. Additionally, since the BWR pressure-suppression design was sensitive to relatively small bypass leakage, the staff position was that the containment should have a steam bypass capability consistent with SRP Section 6.2.1.1.C. This was identified as an open issue.

EPRI was unable to formulate requirements that are adequate to ensure that passive plant vendors' designs that meet them would also meet NRC requirements, criteria, and guidance. The staff will review steam bypass capability during its review of an individual application for FDA/DC. Therefore, this open issue is closed.

7.3 Equipment Design Requirements

Section 7.3.8 of Chapter 5 of the passive Requirements Document requires that the materials for the suppression pool and passive safety injection (PSI) pools and all surfaces that directly come in contact with pool water during normal operation will be selected with corrosion-resistant properties that will ensure that specified water quality limits can be practically maintained. Structures and equipment in the pools will be designed to reduce crud traps. In addition, all material surfaces and protective coatings within the confines of the containment will be able to tolerate the environmental conditions that exist during normal operation or that result from the design-basis accident without producing products that could significantly worsen the effects of the design-basis accident.

The staff considers these requirements acceptable because corrosion-resistant materials and protective coatings that are compatible with the pool water will be used in the construction of suppression and PSI pools.

7.4 Passive Containment Cooling System (BWR)

In Revision 4 of the passive Requirements Document, EPRI added Section 7.4 to Chapter 5 to provide requirements for the BWR passive containment cooling (PCC) system.

Performance Requirements

Section 7..4.2 of the Requirements Document requires that the passive containment cooling system (PCCS) be a safety-grade system capable of maintaining the containment within design pressure limits, assuming an initiating event, the limiting single failure, and no ac power for 72 hours for licensing design basis events. It also requires that PCCS initiation be automatic upon release of significant decay heat into the containment.

The staff considers these requirements to be acceptable provided that the PCCS continues to be available for an extended period of time beyond 72 hours to perform its safety function. This extended operating period can be through the use of onsite facilities or by connecting readily available offsite equipment and resources to replenish the PCCS pool inventory. The basis for the staff acceptance of 72 hours as a design basis is more completely explained in Section 2.2 of Chapter 5 of this report.

System and Equipment Requirements

Section 7.4.3 provides the PCCS system and equipment requirements. Section 7.4.3.1 states that the PCCS "shall consist of a suitable number of redundant components and features." This section also provides requirements for the design to include specific components.

The staff will review the individual design certification packages to determine if adequate redundancy has been provided in the design of the PCCS. Likewise, in regard to which components shall be incorporated into the PCCS design, the staff will review the individual design submittals to determine the adequacy of the system design.

Sections 7.4.3.2 and 7.4.3.3 provide requirements for the PCCS condenser pools. These sections state that the PCCS condenser pool shall be located sufficiently high to allow gravity return flow of condensate to the passive safety injection system (PSIS) pools. In addition, Section 7.4.3.3 of the requirements document states a single pool of water may be used for all the PCCS condensers and associated piping. This section also states that the PCCS condenser pools shall be located outside of the primary containment but inside a structure sufficient to protect a safety-related system. The pools shall be vented to the atmosphere to protect against overpressurization of the pool structure an the vent shall be designed to prevent excessive moisture carry-over in the vented steam to prevent unnecessary loss of inventory. A related requirement is stated in Section 7.4.3.9 which requires the PCCS to be designed in accordance with ASME Section III in order to qualify the PCCS condenser and associated piping as an extension of containment.

The staff finds these requirements to be acceptable. However, the staff notes that the use of a single pool for all PCCS condensers is only acceptable if adequate separation is provided between systems to prevent a common mode failure, and if, it can be shown that a failure cannot occur which could cause a complete loss of inventory. The staff will review the vendor design submittals to determine if the design provides adequate protection against these types of failures. In addition, the location of the PCCS pool outside of containment is only acceptable if the vendor can demonstrate that the PCCS if an acceptable extension of containment. The staff also notes that these requirements are primarily representative of one design and are not the only way to comply with the regulations. While the staff may find them to be acceptable, the staff's finding should not be construed as meaning that this is the only acceptable design for a BWR PCCS.

Section 7.4.3.6 states that vents shall be provided to remove non-condensibles from the system. The staff notes that the non-condensible gases will tend to accumulate in the PCCS condenser, therefore, these vents are necessary because non-condensible gases can inhibit heat transfer. The staff, therefore, finds this acceptable. However, the staff notes that the requirements document does not state where these vents discharge. The staff will review vendor submittals to ensure that adequate measures are taken to prevent the release of fission products to the atmosphere.

Section 7.4.3.7 states that the PCCS lines to the heat exchangers shall be sloped, without pockets, and have the system high point located at the PCCS condenser. This requirement will prevent the condensate buildup in the lines from causing blockages which could degrade heat exchanger performance. Therefore, the staff finds this acceptable.

Section 7.4.3.8 provides the requirements for PCCS instrumentation. The adequacy of instrumentation cannot be determined from the level of detail provided in the Requirements Document. Therefore, the staff will review the

individual design certification packages to determine the adequacy of the PCCS instrumentation and controls.

8 PWR MITIGATION/CONTAINMENT REQUIREMENTS

8.1 Definition

Section 8 of Chapter 5 to the passive Requirements Document provides the performance and equipment design requirements for the PWR primary containment and related systems for the ALWR passive plant.

8.2 Primary Containment

Section 8.2.1.2 of Chapter 5 states that the PWR containment will be designed to provide a leaktight barrier to prevent uncontrolled release of radioactivity in the event of a postulated accident. The primary containment structure will be a large, dry containment with cylindrical steel pressure vessel and a reinforced-concrete secondary shield building. The containment design pressure and temperature must be equal to or greater than the pressure and structural member temperature conditions resulting from postulated loss-of-coolant or steamline or feedwater line break accidents. The containment structure will be designed with sufficient margin to provide severe-accident protection. Best-estimate analyses will be performed to ensure that adequate severe-accident margin exists in the design. The staff will review the adequacy of containment heat removal capability when detailed design analysis data are available from plant designers.

The requirement that the containment be designed to ensure that adequate protection exists from external pressure conditions that may result (e.g., from inadvertent operation of the passive containment cooling system) conforms with the guidance in SRP Section 6.2.2 "Containment Heat Removal System," and, therefore, is acceptable.

Section 8.2.3.2 of Chapter 5 of the passive Requirement Document requires that instrumentation be provided to monitor conditions within the containment during operation and following an accident. This instrumentation originally included measurement of containment radioactivity, pressure, temperature, and in-containment refueling water storage tank (IRWST) level. In response to a staff request for additional information, EPRI added hydrogen (or oxygen) monitoring instrumentation on the list in Revision 2 of the passive Requirements Document. With this change, the staff finds that the requirement for containment monitoring instrumentation conforms with the NRC guidance in SRP Section 6.2.1.1.A, "PWR Dry Containments, Including Subatmospheric Containments," and, therefore, is acceptable.

8.3 Passive Containment Cooling System

Section 8.3 of Chapter 5 describes the requirements for the passive containment cooling system (PCCS). The PCCS will be a dedicated safety system designed to remove heat from the primary containment atmosphere.

Section 4.9 of Appendix B to Chapter 1 of the passive Requirements Document states that the PCCS will be a passive system that will use gravity and natural circulation as driving forces to reduce containment pressure and temperature following a loss-of-coolant accident. Two valve groups will be used to allow water flow by gravity from the PCCS water storage tank to the

top center of the containment shell. Each valve group will contain two valves in series. The staff has identified the reliability of these valves to perform their intended safety function as very important.

The PCCS water storage tank will have sufficient capacity to supply cooling water for 72 hours. Once the PCCS water storage tank is depleted after 72 hours, containment cooling by natural circulation of air will be sufficient to prevent containment pressure from exceeding the design limit. Cooling capability will be evaluated using the licensing design-basis analysis method and assuming the limiting single active component failure. The PCCS will be designated as seismic Category I and Safety Class 3.

In Appendix B to Chapter 1 of the passive Requirements Document, EPRI committed to comply with SRP Section 6.2.1, "Containment Functional Design"; SRP Section 6.2.1.1.A, "PWR Dry Containment, Including Subatmospheric Containments"; SRP Section 6.2.1.3, "Mass and Energy Release Analysis for Postulated LOCAs"; SRP Section 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures"; SRP Section 6.2.1.5, "Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies"; and GDC 16, "Containment Design." In addition, EPRI has committed to comply with the intent of GDC 38, "Containment Heat Removal," and SRP Section 6.2.2, "Containment Heat Removal Systems."

The staff concludes that the basic requirements for the PCCS are acceptable. However, without specific design details, the staff was unable to determine whether passive plant vendors' designs meeting EPRI's requirements would also comply with NRC requirements, criteria, and guidance. Therefore, it will review the adequacy of containment heat removal capability when detailed design analysis data are available from plant designers. The staff's review of valve reliability is given in Chapter 1 of this report.

In its July 22, 1991, letter, EPRI committed to and subsequently added to Section 8.3.3.11.2 of Chapter 5, a provision requiring the PWR PCCS to be actuated on loss of air or loss of dc power.

Chemistry Control

Section 8.3.3.3 of Chapter 5 specifies that the designer will provide means to permit sampling of the water contained in the system during normal plant operation to adjust water chemistry, including mixing. The staff concludes that this requirement does not conflict with Commission regulations.

Maintenance and Inspection

Section 8.3.3.12 of Chapter 5 specifies that the PCCS will be designed to facilitate maintenance, including inservice inspection, in accordance with the general requirements in Section 8 of Chapter 1 of the passive Requirements Document. The staff concludes that these requirements do not conflict with Commission regulations.

9 CONCLUSION

The staff concludes that the EPRI requirements established in Chapter 5 of the passive Requirements Document for the design of engineered safety systems do not conflict with current regulatory guidelines and are acceptable. However, by themselves, they do not provide sufficient information for the staff to determine if the engineered safety systems design will be adequate. Applicants referencing the passive Requirements Document will be required to demonstrate compliance with the additional guidance in the SRP or provide justification for alternative means of implementing the associated regulatory requirements.

Therefore, the staff concludes that Chapter 5 specifies requirements that, if properly translated into a design in accordance with the NRC regulations in force at the time the design is submitted, should result in a nuclear power plant whose engineered safety systems will perform as designed and have all the attributes required by the regulations to ensure that there is no undue risk to the health and safety of the public or to the environment. In addition to complying with existing regulations, such a facility would also be consistent with Commission policies on severe-accident protection and public safety goals.

APPENDIX A
DEFINITIONS AND ACRONYMS

Appendix A of Chapter 5 of the passive Requirements Document contains definitions of terms and acronyms. The staff has provided a consolidated list of acronyms in Volume 1 of this report.

APPENDIX B
DEFINITION OF PHYSICALLY BASED SOURCE TERM FOR THE PASSIVE PWR AND PASSIVE BWR

Appendix B to Chapter 5 of the passive Requirements Document originally contain EPRI's definitions for the physically based source terms for the passive PWR and the passive BWR. In Revision 5, EPRI deleted this appendix.

CHAPTER 6, "BUILDING DESIGN AND ARRANGEMENT"

1 INTRODUCTION

This chapter of the SER documents the NRC staff's review of Chapter 6, "Building Design and Arrangement," of the passive Requirements Document through Revision 5. Revision 5 was submitted to the NRC by letter dated October 1, 1993. Chapter 6 was prepared, under the project direction of EPRI and the ALWR Utility Steering Committee, by Bechtel Power Corporation; Commonwealth Edison Company; Duke Power Company; General Electric Company; MPR Associates, Inc.; S. Levy Incorporated; Sargent and Lundy Engineers; Stone and Webster Engineering Corporation; Westinghouse Electric Corporation; and EPRI.

1.1 Review Criteria

Section 1 of Volume 1 of this report describes the review criteria used by the staff during its review of Chapter 6 of the passive Requirements Document

1.2 Scope and Structure of Chapter 6

Chapter 6 of the passive Requirements Document defines the ALWR Utility Steering Committee's requirements for the design and arrangement of the buildings for a facility where passive safety systems will be used. The key topics addressed in Chapter 6 include plant arrangement, which covers building; piping; electrical raceway; heating ventilating, and air conditioning ductwork and equipment layout; configuration; interface and support requirements; and arrangement of features to accommodate all stages of plant life from construction and erection through testing, including operation, maintenance, repair, and replacement.

1.3 Policy Issues

During its review of Chapter 6 of the passive Requirements Document, the staff did not identify any issues that involve policy questions for the technical areas discussed in this chapter, other than those already identified in Commission papers listed in Appendix B to Chapter 1 of this report.

1.4 Outstanding Issues

The DSER for Chapter 6 of the passive Requirements Document contained the following outstanding issues:

Open Issues

P.6.0-1 classification of non-safety-grade auxiliary systems as vital equipment (2.3.3)

P.6.0-2 turbine-generator building safe shutdown earthquake loading conditions (4.4)

P.6.O-3 location of the control complex (4.5.4)

P.6.O-4 requirements and acceptance criteria for human factors considerations (4.5.4)

Confirmatory Issues

None.

The final disposition of each of these issues is discussed in detail in the appropriate section of this chapter, as indicated by the parenthetical notation following each issue. All outstanding issues identified in the DSER for Chapter 6 have been resolved.

1.5 Vendor- or Utility-Specific Items

The DSER for Chapter 6 contained the following vendor- or utility-specific items. The final disposition of each item is discussed in the appropriate section of this chapter, as indicated by the parenthetical notation following each item.

P.6.V-1 method for inspecting structural degradation (2.1.1)

P.6.V-2 evaluation of the engineering backfill (2.1.1)

P.6.V-3 use of American National Standards Institute (ANSI) 10.4-1987 (2.1.1)

P.6.V-4 structural modules to be used in construction (2.2)

P.6.V-5 redundant non-safety-grade auxiliary systems within the plant protected area (2.3.3)

P.6.V-6 review of site-unique security and contingency plans (2.3.6)

P.6.V-7 programs for controlling and storing toxic materials (2.3.7)

P.6.V-8 shielding design requirements, shielding computer codes, and radioactive material sources (2.4)

P.6.V-9 use of ANSI/American Institute of Steel Construction (AISC) N-690 (4.1.2 and 4.1.3)

P.6.V-10 evaluation of potential high-radiation areas (4.1.7)

P.6.V-11 design of the common basemat (4.1.9)

P.6.V-12 reinforced-bar design criteria for vinyl-coated rebars (4.1.9)

P.6.V-13 material control provisions inside the containment (4.2.2 and 4.2.3)

P.6.V-14 control of access to the reactor containment (4.2.2 and 4.2.3)

P.6.V-15 design features that preclude potentially lethal radiation levels (4.2.2)

P.6.V-16 containment design details for aerosol and radioactive gases (4.2.3)

P.6.V-17 use of ANSI/American Society of Mechanical Engineers (ASME) NOG-1-1983 (4.2.3)

P.6.V-18 design of the control complex (4.5.4)

The following vendor- or utility-specific items remain and will be addressed during the staff's review of an individual application for FDA/DC. The staff's position on each item is provided in the appropriate section of this chapter, as indicated by the parenthetical notation following each item.

P.6.V-1 method for inspecting structural degradation (2.1.1)

P.6.V-2 effect of modules in, overall structural configuration (2.1.1 and 2.2)

P.6.V-3 adequacy of engineering backfill (2.1.1)

P.6.V-4 use of ANSI 10.4-1987 (2.1.1)

P.6.V-5 fire hazard analysis (2.3.1)

P.6.V-6 sabotage vulnerability analysis (2.3.3)

P.6.V-7 regulatory treatment of non-safety systems (2.3.3)

P.6.V-8 site-unique security and contingency plans (2.3.6)

P.6.V-9 programs for controlling and storing toxic materials (2.3.7)

P.6.V-10 shielding design requirements, shielding computer codes, and radiation sources (2.4 and 4.1.6)

P.6.V-11 use of ANSI/AISC N-690 (4.1.2 and 4.1.3)

P.6.V-12 potential high-radiation areas (4.1.7 and 4.2.2)

P.6.V-13 design of the common basemat (4.1.9)

P.6.V-14 design criteria for vinyl-coated rebars (4.2.1)

P.6.V-15 control of access to the reactor containment (4.2.2 and 4.2.3)

P.6.V-16 use of ANSI/ASME NOG-1-1983 (4.2.3)

P.6.V-17 structural integrity of turbine building (4.4)

P.6.V-18 control room habitability (4.5.4)

P.6.V-19 design of technical support center (4.5.5)

2 POLICY STATEMENTS AND KEY REQUIREMENTS

2.1 Policy Statements

Section 1 of Chapter 6 of the passive Requirements Document contains those policies established by the ALWR Utility Steering Committee concerning constructibility, operability, maintainability, radiological sabotage, and standardization as they relate to building design and arrangement. The following is a summary of these policies:

- With regard to the arrangement of buildings, emphasis will be placed on improved constructibility through material selection, component standardization, and space layout to minimize interferences and improve accessibility.

- Operability and maintainability will be improved through activities that emphasize accessibility in the design, including the use of early periodic design reviews and the separation of potentially radioactively contaminated spaces from noncontaminated spaces.

  EPRI states that operation and maintenance will be improved by providing space for access to operate and maintain components (space requirements will be determined by performing maintenance reviews at early design stages), and by separating potentially contaminated areas from noncontaminated areas. Historically, plant maintenance has accounted for most of the occupational dose at light-water reactors. Increasing accessibility to components located in radiological control areas (RCAs) will facilitate maintenance operations on these components, will decrease the maintenance time spent in the RCAs, and will result in overall lower occupational doses.

- Vulnerability to radiological sabotage will be minimized through consideration of plant arrangements and systems design enhancements, access control, and implementation of the physical security provisions as described in Chapter 9 of the passive Requirements Document.

- The standardization objectives outlined in Chapter 1 will be implemented through the use of a standard power generation complex and a reference site arrangement as described in Chapter 6.

2.1.1 General Design Requirements

Section 2.1 of Chapter 6 specifies the general design requirements related to (1) human factors, (2) optimization of plant volume, (3) architecture, (4) standardization, (5) structural design basis, (6) plant life, (7) site envelope, and (8) design process.

Human Factors

Section 2.1.1.2 of Chapter 6 states that equipment located in normally inaccessible areas (including areas in which the dose rate exceeds 100 mrem/hr) will be limited to those items that do not require surveillance testing during operation, that are extremely reliable, or that will not impair

plant availability if they fail. This limitation ensures the accessibility of equipment that may require frequent maintenance or surveillance; therefore, this requirement is acceptable.

Structural Design Bases

Section 2.1.5 of Chapter 6 requires that plant structures be designed according to the general design bases specified in Chapter 1, Section 4, supplemented by the specific requirements of Chapter 6.

Section 2.1.5.2 of Chapter 6 states that appropriate and achievable construction tolerances will be provided for dimensions, locations, and clearances for all structures, systems, and components. The tolerances will be included in installation specifications and on drawings. The following paragraphs document EPRI's resolution of the staff concerns regarding the structural design-bases requirements in Chapter 6.

(1) The staff was concerned that Section 2.1.5.2 of Chapter 6 did not address the effect of thermal growth of structural steel members due to fire hazards and the design-basis loss-of-coolant accident (LOCA). The staff, however, found that the requirements in Sections 4.5.5.1.5 and 4.5.5.6 of Chapter 1 adequately addressed the issue of a design-basis LOCA and in-plant fire hazards, respectively. Therefore, the staff requested that Section 2.1.5.2 of Chapter 6 be revised to reference these two sections in Chapter 1. EPRI implemented this staff request in Revision 3 of Section 2.1.5.2, and this concern is considered resolved.

(2) The staff was also concerned about plant arrangements of all safety-related structures, both steel and concrete, with respect to accessibility for inspection and surveillance. In particular, the staff was concerned about a reliable means for assessing the potential for structural degradation if direct inspections are excluded (e.g., where filler material is placed between the containment and adjacent concrete structures). Section 4.2.11.1.2 of Chapter 6 requires a common basemat for the power generation complex (PGC) structures, including the control, auxiliary, and containment buildings, which permits access to both sides of all interior structural walls. In the case of pressurized-water reactor (PWR) plants, Sections 4.3.4.1.1 and 4.3.4.1.2 were added to Chapter 6; these specifically require access outside and inside the steel shell containment for inspection. Requirements in these sections were sufficient to address the staff concern regarding all structural walls within the common basemat and those portions of exterior walls along the edge of the common basemat that will not be immediately adjacent to another building. Requirements were insufficient, however, in addressing the issue regarding exterior walls along the edge of the common basemat that are immediately adjacent to the exterior walls of another building supported on a separate foundation (e.g., between the exterior walls of control and turbine building). In its September 29, 1992, submittal, EPRI stated that the exterior walls of the seismic Category I structures will be constructed of reinforced concrete. Because the reinforcing steel will be embedded with a minimum of 2 inches of concrete cover and typically the exterior face will be covered by a waterproof membrane, there should be no need to inspect the

exterior face of walls nor the two foundations for structural degradation. On the basis of (1) engineering judgment, (2) the fact that it is very difficult to have access for a complete inspection of the foundations between buildings after the construction is completed and the backfill is in place, and (3) the Section 4.2.11.6.4 requirements for using epoxy-coated reinforcing steel in corrosive environments, the staff concludes that a complete inspection of the exterior embedded walls and foundations may not be necessary. However, plant designers should provide access or other means for localized inspection of structural degradation of the walls and foundations. The staff will review the adequacy of design and procedures for such localized inspection during its review of an individual application for FDA/DC.

(3) The third staff concern was related to stiffness degradation under repeated loads in modularly constructed concrete structures and differences between modularly and conventionally constructed concrete structures. In addition, the staff was concerned about the effects of such differences on structural integrity and piping and equipment design. In a letter dated November 18, 1991, EPRI stated that safety-related concrete structures in ALWR plants will be constructed primarily of monolithic concrete similar to past practice because such construction is more cost effective than making modules out of precast concrete, and modular construction will be used for equipment modules, including piping and electrical systems, supported by a structural steel framework attached to the primary building at connection points. EPRI maintained that the seismic response of modules will not be substantially different from current plants and, hence, a justification for differences in stiffness was unwarranted. The staff finds that this position does not completely address the issue of differences between modularly and conventionally constructed concrete structures. In its review of an individual application for FDA/DC, the staff will ensure that the use of precast concrete modules does not lead to an overall structural configuration that can lead to an unacceptable structural design.

(4) The fourth staff concern was about the lack of emphasis in Chapter 6 on anchorage design and installation for safety-related tanks. Section 12.6.2 of Chapter 1, however, requires that anchor bolts for flat-bottom be extended above the foundation and that anchor bolt chairs be extended full height and welded continuously to develop ductility. Because this requirement will permit stresses from a seismic event to be transferred from the tank to the foundation without rupturing, the staff concludes that it is acceptable.

(5) The fifth staff concern was that Chapter 6 did not fully address the potential for steel containment corrosion, spent fuel pool leakage, and degradation of intake structures. The staff determined, however, that Chapter 1, Section 4.3.4.1.1, requires the inclusion of appropriate corrosion allowance in the design of steel shell containments and Chapter 7, Section 2.3.1.1.6, requires a leak chase system be installed to collect potential leakage behind the weld seams of the spent fuel pool liner plate. Therefore, this concern is considered resolved.

With the exception of the vendor- or utility-specific items identified above, the staff concludes that the structural design-related requirements in Section 2.1 of Chapter 6 of the passive Requirements Document are consistent with staff regulatory positions and are, therefore, acceptable.

Site Envelope

Section 2.1.7 of Chapter 6 requires that the founding material with the properties specified in Table 1.2-6 of Chapter 1 be assumed to occur at a depth of 25 to 30 feet below average site grade level. The significance of requiring a 25- to 30-foot standard embedment depth was not clear. In addition, it was not clear how the plant designer would address the adequacy of a shallow soil site founded on bedrock, which is more competent than the material required in Table 1.2-6 of Chapter 1.

In a letter dated November 18, 1991, EPRI stated that, by definition, standard plants will have a standard embedment and the standard embedment of 25 to 30 feet for nuclear island structures was originally specified in order to achieve an appropriate bearing capacity at most soil sites. If sites have competent rock at shallower depth, the rock will be excavated to the depth that accommodates the standard plant arrangement to permit major access openings to be at grade level. Because this requirement is included in the passive Requirements Document to facilitate equipment access and because there is no conflicting regulatory guidance or requirements in this area, the staff concludes that it is acceptable.

In its request for additional information dated April 24, 1991, the staff asked EPRI to address the following three issues if the embedment of 25 to 30 feet will be accomplished through engineering backfill:

- selection of backfill material, and quality control and quality assurance to ensure the reliability of the backfill
- location of seismic input
- description of soil characteristics for soil-structure interaction (SSI) analysis

In its response dated July 2, 1991, EPRI stated that it is the responsibility of the plant designer to establish these parameters under the guidance provided in Standard Review Plan (SRP) Sections 3.7.1 to 3.7.3, with which the ALWR program is committed to comply. This response is acceptable, and the staff concerns related to embedment depth are resolved. The staff will evaluate the adequacy of the engineering backfill during its review of an individual application for FDA/DC.

Design Process

Section 2.1.8.1 of Chapter 6 requires the use of "proven and verified" computer programs for design process analyses. The staff was concerned about the criterion "proven and verified" and the qualification criteria and the basis thereof. EPRI, in a letter dated November 18, 1991, stated its position

that ANSI 10.4-1987 be used as the guidance for verifying and validating computer programs and this ANSI standard was referenced in Table 1.4-2, Chapter 1. The staff position is to review the acceptance of this ANSI standard for consistency with the regulatory position. Although ANSI 10.4-1987 is included in the list of industry technical standards shown in Table 1.4-2, Chapter 1, this standard has not been endorsed by the staff. As discussed in Section 4.4 of Chapter 1 of this report, instances where the plant designer uses codes and standards that conflict with NRC positions must be resolved with the NRC staff and fully documented during the review of an individual application for FDA/DC.

In addition, Section 6.1 of Chapter 10 of this report, the staff provides a detailed evaluation of the overall requirements for the MMIS software. More specifically, Sections 6.1.2.6 and 6.1.2.7 of Chapter 10 provide requirements for a software verification and validation program.

2.2 Design for Construction

Section 2.2 of Chapter 6 of the passive Requirements Document identifies the key requirements necessary to ensure that the plant construction is consistent with operational and maintainability goals. Such key requirements include (1) construction sequence, (2) segregation of safety-related and non-safety-related construction areas, (3) modularization, and (4) startup testing. The staff reviewed the structural design aspect of the modularization requirement as stated in Section 2.2.4 of Chapter 6.

Because modular construction is proposed throughout the passive Requirements Document even though it has not been widely used in the construction of safety-related structures in nuclear power plants, the staff concluded that an analysis should be performed to demonstrate that the same degree of structural strength and reliability provided in conventional nuclear plant construction will be maintained with the proposed modularization scheme. In a letter dated November 18, 1991, EPRI stated that any unusual structural modules that could perform other than as an integral or composite part of the structure will require justification on a plant-specific basis. The staff concludes that EPRI's position is acceptable, and the staff considers the concern resolved. The staff will evaluate any unusual structural modules used during its review of an individual application for FDA/DC.

2.3 Design for Safety

2.3.1 Fire Protection

Section 2.3.3 of Chapter 6 of the passive Requirements Document specifies the following requirements for passive fire protection features:

- Construction materials and the fire loading on buildings will conform to SRP Section 9.5.1, "Fire Protection Program" (Branch Technical Position CMEB 9.5-1, July 1981).
- Cooling towers will be constructed of noncombustible materials.
- Fire barriers and openings through fire barriers will conform to SRP Section 9.5.1.

- Computers that are not part of the control room complex and that will perform safety-related functions will be separated from other areas of the plant and from their redundant backups by 3-hour-fire barriers.

- Switchgear rooms containing safety-related equipment will be separated from the remainder of the plant by 3-hour-fire barriers, redundant divisions will be separated from each other by 3-hour-fire barriers, and all equipment will be accessible, to the extent practicable, on all sides for manual fire suppression.

- Redundant safety-related panels remote from the control room complex will be separated from each other by 3-hour-fire barriers, and panels that provide remote shutdown capability will be separated from the control room complex by 3-hour-fire barriers.

- Safety-related battery rooms will be separated from each other and from other areas of the plant by 3-hour-fire barriers, and dc switchgear and converters will not be located in the battery rooms.

- Exterior walls of safety-related buildings that may be exposed to fire hazards will be designed as 3-hour-fire barriers, and the openings and penetrations will be sealed with materials having a 3-hour-fire rating. Outdoor oil-filled transformers will have features for confining oil spill and water deluge or they will drain away from buildings. Exterior walls located in the vicinity of transformers will have a 3-hour-fire rating, and openings in the walls will be avoided.

- Diesel fuel oil storage areas (except day tanks) will not be located inside buildings containing safety-related equipment. Tanks will be totally buried or will be located above ground at least 50 feet from any building containing safety-related equipment or will be housed in a separate building with a 3-hour-fire rating.

- Personnel access and escape routes will be provided for each fire area. Fire exit routes will be clearly marked. Stairwells outside the primary containment serving as escape routes, access routes for fire fighting, or access routes to safe-shutdown equipment will be enclosed in masonry or concrete towers, with a minimum 2-hour-fire rating, and will be provided with self-closing Class B fire doors and emergency lighting. Fire exits will be provided in accordance with the requirements of Chapter 5 of National Fire Protection Association (NFPA) 101, "Life Safety Code." Stairwell design will conform to Chapters 6 and 7 of NFPA 803, "Fire Protection for Light Water Nuclear Power Plants."

- Floor drains big enough to remove water used for fire fighting without flooding safety-related equipment will be provided in those areas where fixed water fire suppression systems will be installed. Floor drains will also be provided in other areas where hand hoses may be used, if such water supply could cause unacceptable damage to safety-related equipment. Where gas fire suppression systems will be installed, drains will be provided with adequate seals or the suppression system will be of such size as to compensate for the loss of the suppression agent

through the drains. Drains in areas containing combustible liquids will have provisions for preventing the backflow of combustible liquids into safety-related areas through the interconnected drain systems.

- The control complex will be separated from the remainder of the plant by 3-hour-fire barriers at walls, ceiling, and floor. Peripheral rooms in the control complex will be separated from the control room by noncombustible construction with a fire-resistance rating of 1 hour.

The staff evaluated EPRI's requirements for the fire protection system in the passive Requirements Document against the criteria of SRP Section 9.5.1 and supplemental guidance issued by the Commission. Three examples of such supplemental guidance are (1) Generic Letter (GL) 81-12, which contains information on safe-shutdown methodology; (2) GL 86-10, which contains important technical information, such as that pertaining to conformance with NFPA codes and standards; and (3) the Commission's staff requirements memorandum on SECY-90-016 dated July 26, 1990. The enhanced fire protection criteria and the basis for its use are discussed in Section 3 of Chapter 9 of this report.

An ALWR designer or applicant will be required to identify any deviations from these codes and standards and to describe in the fire hazard analysis for an individual application for FDA/DC, the deviations and measures taken to ensure that equivalent protection is provided. EPRI commits in Chapter 9 to design, install, and test the plant fire protection system in accordance with SRP Section 9.5.1. The staff understands that the EPRI commitment is acceptable, provided designers or applicants also meet other applicable guidance contained in GL 81-12, Revision 1, and GL 86-10. This commitment is acceptable.

The staff's evaluation of the fire protection requirements specified in Chapter 6 of the passive Requirements Document appears below:

Fire Barrier Design

EPRI has stated its intention to conform to the fire barrier qualification guidance contained in SRP Section 9.5.1. This commitment is acceptable.

Oil-Filled Transformers

Section 2.3.3.11 of Chapter 6 requires that outdoor oil-filled transformers will have oil-spill and water-deluge confinement features or drainage away from the buildings. Such transformers will be located at least 50 feet from buildings. Where oil-filled transformers will be located within 50 feet of buildings, the exterior walls in the vicinity of the transformers will have a fire-resistance rating of at least 3 hours, and wall openings will be avoided. The drainage system will confine or direct potentially flaming oil from a transformer fire so that other electric power supply circuits will not be affected.

This requirement is consistent with the enhanced fire protection criteria discussed in Chapter 9 of this report and is, therefore, acceptable.

Fire Exits

Section 2.3.3.13 of Chapter 6 requires that fire exits be provided in accordance with Chapter 5 of NFPA 101, "Life Safety Code." This meets the criteria provided in the review guidance of SRP Section 9.5.1 and is acceptable.

Heating, Ventilating, and Air Conditioning Design

Section 8.2.1.1.17 of Chapter 9 requires that for fire zones protected with gas fire suppression systems, the zone heating, ventilating, and air conditioning HVAC system will be interlocked with the gas fire suppression system in accordance with NFPA 12, "Carbon Dioxide Extinguishing Systems." This commitment is acceptable since it meets the criteria provided in the review guidance of SRP Section 9.5.1. Additional guidance in this area is also given in ANSI/American Nuclear Society (ANS) 59.2-1985, "Safety Criteria for HVAC Systems Located Outside Primary Containment." (ANS is planning minor revisions to update this standard.)

Partitions Inside the Control Complex

Section 2.3.3.15 of Chapter 6 requires that peripheral rooms in the control complex be separated from the control room by a noncombustible structure with a fire-resistance rating of 1 hour in accordance with Position 7.b of SRP Section 9.5.1.

The staff's evaluation of EPRI's requirements for fire protection systems is provided in Section 3 of Chapter 9 of this report.

2.3.2 Postaccident Access

Section 2.3.4 of Chapter 6 requires that the plant designer prepare a plan for postaccident access for surveillance and recovery operations and that, in accordance with the requirements resulting from the accident at Three Mile Island (TMI), the plant arrangement include provisions for postaccident access for these operations. The plan to be prepared by the plant designer, as required by EPRI, will identify areas requiring postaccident access. These areas must be shielded so that required operations in these areas are not adversely affected by postaccident radiation levels. The systems designed to function after an accident may include, but are not limited to, the containment system, the residual heat removal system, the core spray system, the reactor core isolation cooling system, the postaccident sampling system, and the standby gas treatment system. Vital areas that may require personnel access after an accident include the control room and technical support center (continuous occupancy), the remote shutdown panel (frequent occupancy), and the sampling station and sample analysis area (infrequent occupancy). Each applicant for FDA/DC must perform a design review of the plant shielding to ensure that the plant's vital areas are accessible after an accident in accordance with the criteria of TMI Action Plan Item II.B.2 (NUREG-0737).

2.3.3 Security Requirements

Section 2.3.5 of Chapter 6 specifies the general security requirements related to building design and arrangement. EPRI states that the design will (1) minimize the security and radiological control points to the extent practicable, (2) enhance the capability to resist radiological sabotage, and (3) not hinder necessary operator access during emergencies.

Section 2.3.5.1 of Chapter 6 requires the plant designer to evaluate access to vital and protected areas during plant construction, normal operation, refueling operations, and outages. (Section 5.2.4.2 of Chapter 9 also requires an evaluation of security system impact on plant operations, maintenance, and testing.) Section 2.3.5.2 of Chapter 6 requires that building exteriors be arranged to avoid irregular shapes to minimize the burden of complying with security requirements. Section 2.3.5.3 requires the plant designer to define emergency paths from the control room to the remote shutdown panel(s), the technical support center, and the on site power supply. It also specifies that security system failures should not result in the blockage of operator emergency access.

Section 2.3.5.4 of Chapter 6 originally specified that, to the extent practical, plant arrangement and building design features will be used to help protect against insider sabotage. The rationale for this section stated that this requirement, coupled with appropriate administrative procedures, will eliminate the possibility that a badged individual could reach and enter all divisions of safety equipment within a short time. In its letter dated April 17, 1991, the staff expressed concern that, despite the provisions of Sections 2.3.5.1 and 2.3.5.3, Section 2.3.5.4 appeared to have the potential for delaying emergency access to vital equipment and might provide little additional benefit to protection against insider sabotage. In its June 24, 1991, letter, EPRI committed to change Chapter 6, Sections 1.5.2 and 2.3.5, and Chapter 9, Section 5.2.4.1, to require security access controls to vital areas only at vital area boundaries and not between redundant divisions of vital components. The staff has confirmed that these changes have been made and resolve its concerns about the possibility of security system interference with safe plant operation. Protection against insider sabotage still will be provided by vital area access controls and by passive plant design features that reduce opportunities for radiological sabotage. Additional protection against insider sabotage may result from the vulnerability analysis required in Chapter 9. Further protection against insider sabotage will be provided by access authorization and fitness-for-duty programs required by 10 CFR Parts 73 and 26.

In response to staff comments, EPRI revised Section 2.3.5.5 of Chapter 6 to consolidate requirements dealing with vital areas and barriers for vital safety equipment in Chapter 9 of the passive Requirements Document rather than in Chapter 6.

Chapter 3 requires redundant non-safety-grade auxiliary systems to be capable of maintaining reactor coolant inventory and removing decay heat for transients and small LOCAs without challenging the vital safety systems (except for the reactor protection system). These non-safety-grade auxiliary systems are not identified as vital equipment in Chapter 9. The location of the service water systems, may be important for a non-safety-grade auxiliary

system, such as the BWR reactor water cleanup (RWCU) system, to be effective in providing additional defense-in-depth protection against sabotage. However, evaluating the effectiveness of the redundant non-safety-grade auxiliary systems in providing additional defense-in-depth sabotage protection can be deferred until the staff reviews an individual application for FDA/DC and the sabotage vulnerability analysis that Chapter 9 of the passive Requirements Document requires.

Nothing in Section 2.3.5 of Chapter 6 would prevent compliance with existing NRC security requirements. In the DSER for Chapter 6, the staff stated that the Commission had not yet defined the regulatory treatment of non-safety systems for passive plant designs. Because the resolution of this issue could result in some non-safety-grade auxiliary systems being identified as deserving special attention, making it appropriate to protect them as vital equipment, this was identified as an open issue in the DSER. In its letter dated May 26, 1993, EPRI proposed an approach to determine which systems should receive which kind of regulatory oversight. A key element of this process is to ensure that the passive plant vendors use probabilistic risk assessment methods to examine reliability of the safety and non-safety systems and ensure that important risk insights are considered in the design. This process, and the staff's evaluation of it, are described in greater detail in SECY-94-084, which is provided in Annex B of Appendix B to Chapter 1 of this report. The staff will review implementation of EPRI's process during its review of an individual application for FDA/DC to ensure that the passive plant vendors follow this process satisfactorily. Therefore, this open issue is closed.

2.3.4 Internal Flooding Requirements

Section 2.3.6 of Chapter 6 specifies the following requirements related to internal flooding:

- Plant arrangements will be such that line breaks in one division of a non-safety-related system will not cause flooding in areas housing equipment of another division of the same system or of another system of another division.

- Compartments subject to flooding or pressurization as a result of an accident or equipment failure will be identified, and design criteria will be established to maintain the structural integrity of the compartment.

- The effects of flooding due to potential breaks of non-safety-related systems, such as the circulating water or service water systems, will be considered in locating electrical equipment in the auxiliary, control, or turbine buildings.

- Cubicles containing radioactive tanks will be designed to retain the contents of the largest tank in the cubicle to prevent the spread of radioactive fluid in the event of a tank rupture.

The design requirements in Section 2.3.6 of Chapter 6 for protection against internal flooding are consistent with the guidance in SRP Section 3.4.1, "Flood Protection," and Regulatory Guide (RG) 8.8 regarding personal exposure and are, therefore, acceptable.

2.3.5 Severe-Accident Considerations

Severe-accident considerations supplementing those of Chapter 5 are contained in Sections 2.3.7 and 4.3 of Chapter 6 of the passive Requirements Document and are discussed by the staff in Section 4 of this chapter. The staff's evaluation of EPRI's severe-accident considerations appears in Chapter 5 of this report.

2.3.6 Protection From External Threats

Requirements for the analysis of manmade hazards and of natural phenomena are given in Chapter 1 of the passive Requirements Document. The staff's evaluation of EPRI's requirements related to protection against flooding from external causes appears in Chapter 1 of this report. The discussion of protection against sabotage in Section 2.3.8 of Chapter 6 of the passive Requirements Document is taken by the staff to mean only that sabotage need not be included in site-specific probabilistic risk analyses. The requirement for the plant designer analyses to determine if any other plant design features are practicable for protection against sabotage in addition to those specified in Chapters 5 and 6 is established in Chapter 9. In addition, acceptable site-unique security and contingency plans required by 10 CFR 50.34 will need to be included in an operating license application. The staff will evaluate site-unique plans and analyses during its review of an individual application for FDA/DC.

2.3.7 Toxic Material and Combustible Gases

Section 2.3.9 of Chapter 6 specifies that toxic materials will be stored at least 50 feet from the control room or intake to the compressor of the breathing air system. This will minimize the potential for contaminating the control room atmosphere or respirator breathing air supply in the event of a leak of toxic gas from a tank or from the bottled gas storage area. The minimum distance between the toxic gas source and the control room air intake is dependent on the amount and type of the toxic gas in question, the container size, and the available control room protection provision. Therefore, the staff will review the toxic gas hazards and relative location of the toxic gas sources on a site-specific basis during its review of an individual application for FDA/DC.

2.4 Design for Operation and Maintenance

Section 2.4.1 of Chapter 6 of the passive Requirements Document requires that the plant designer develop a maintainability evaluation that will identify limiting tasks that control the sizes of aisleways, hatches, and lifting devices at the time the plant's general arrangement drawings are being developed. This maintainability evaluation will be used to identify needed work spaces and clearances; the adequacy of platforms and ladders; lifting and pulling points in walls, ceilings, and floors; and access routes to be followed. Aisleways will be wide enough to provide unobstructed pathways for the removal or installation of all planned components (except in-building large tanks). Provisions for lifting and handling major equipment and components weighing more than 50 pounds will be provided in the form of cranes, monorail hoists, and rigging attachment points. All equipment requiring periodic access for operation, maintenance, or inspection will have

access stairs and work platforms. To minimize radiation exposure levels to plant operating personnel, the plant designer will include features in the design of the plant to facilitate the use of robots for plant maintenance activities. These features, implemented as a result of the maintainability evaluation described above, are intended by EPRI to facilitate access during maintenance, operations, and inspection activities. These features, if properly implemented, will help reduce occupational exposures and are in compliance with the guidelines of RG 8.8. Several of these design features (such as the capability to remove all major plant components, except the reactor vessel and some large tanks) will also facilitate eventual decommissioning operations.

The objective of the plant's radiation shielding is to protect operating personnel against radiation, both inside and outside the plant, during normal operation, including anticipated operational occurrences, and during reactor accidents. Section 2.4.4 of Chapter 6 indicates that the plant and building arrangement will be designed to minimize personnel exposure to radiation and contamination by optimizing shielding and equipment maintenance design features. The passive Requirements Document states that source terms used for shielding design will be based on NRC regulations and existing plant surveys and will take into account any reactor coolant water additives. In Revision 2 of Appendix B to Chapter 1, EPRI commits to meet SRP Sections 12.3 and 12.4, which require the listing of acceptable shielding design computer codes used. It is outside the scope of the passive Requirements Document to list specific shielding design computer codes. However, the ALWR designer or applicant will identify all such shielding codes planned for use in shielding design. Because Chapter 6 does not fully describe shielding design requirements that are consistent with SRP Sections 12.3 and 12.4, the staff will ensure that the ALWR designer or applicant complies with this guidance. The ALWR designer or applicant will be required to provide a description of radiation sources, during normal operations and accident conditions in the plant, that will be used as the basis for designing the radiation protection program and for shield design calculations. This description should include isotopic composition, location in the plant, source strength and source geometry, and the basis for the values. The above three items: (1) computer codes for shielding design evaluation, (2) details of shielding design, and (3) a description of radiation sources that is used as the basis for the shielding design are all vendor- or utility-specific issues. EPRI has stated that these issues will be addressed in an individual application for FDA/DC and that the resolution of these issues will be in accordance with the guidance provided in the appropriate sections of the SRP. The staff will determine the acceptability of these resolutions.

3 OVERALL SITE ARRANGEMENT

3.1 Site Drainage

Section 3.3.1 of Chapter 6 of the passive Requirements Document requires that the site drainage capacity be adequate to handle the local probable maximum precipitation (PMP), including runoff from local topography, without flooding the site. The passive Requirements Document originally referenced ANSI/ANS 2.8-1981 as a guide to be used by the plant designer in establishing the PMP and site drainage requirements.

Because the staff had not reviewed the guidance in ANSI/ANS 2.8-1981 for consistency with regulatory positions, in its letter dated April 24, 1991, the staff requested that SRP Section 2.4.10, "Flood Protection Requirements," and GL 89-22, "Resolution of Generic Safety Issue No. 10: Design for Probable Maximum Precipitation," be referenced in the passive Requirements Document in lieu of ANSI/ANS 2.8-1981. In its letter dated July 2, 1991, EPRI revised Section 3.3.1 in Revision 2 to reference SRP Section 2.4.10 as the basis for establishing PMP. Therefore, this EPRI requirement is acceptable.

3.2 Site Layout

Section 3.3.2.5 of Chapter 6 specifies the standard levels of embedment for both the BWR building and the PWR containment structure and auxiliary building. Appendices B and C of Chapter 6 show the typical embedment of about 50 feet for the BWR building and 40 feet for the PWR building. These embedment levels conflict with the 25-foot to 30-foot standard embedment specified in Section 2.1.7.1 of Chapter 6, and in a letter to EPRI dated April 24, 1991, the staff requested an explanation of the difference and the intent of the passive Requirements Document. In its response dated November 18, 1991, EPRI stated that the intent of the requirement in Section 2.1.7.1 is to further clarify the founding material bearing capacity requirement specified in Table 1.2-6 of Chapter 1. EPRI also stated that the standard plant embedment levels specified in Section 3.3.2.5 are based on tradeoffs between founding material bearing capacity, construction cost, and maintenance access and considerations. The EPRI response is adequate to resolve the staff concern raised in its letter dated April 24, 1991.

Section 3.3.2.6 of Chapter 6 originally required that plant grade be established at an elevation higher than the probable maximum flood (PMF) elevation as defined in ANSI/ANS 2.8, including consideration of the effect of wave runup and splash for seaside and lakeside locations. In a letter to EPRI dated April 24, 1991, the staff asked EPRI to replace the reference to ANSI/ANS 2.8 with SRP Section 2.4.10. EPRI agreed to this request in its response dated November 18, 1991, and has implemented its commitment in Revision 3 of Chapter 6. The basis specified in Section 3.3.2.6 for establishing PMF elevation is acceptable and the staff's concern stated in the letter dated April 24, 1991, is resolved.

3.3 Service Facilities

Section 3.3.4 of Chapter 6 specifies the requirements for service facilities. Included within the scope of the passive Requirements Document are requirements pertaining to

- the accommodation of normal operating staff, permanent support staff, contract workers, additional personnel required during outages, regulatory personnel, and visitors
- space dedicated for future expansion
- warehouse facilities

The design of the plant service facilities will be based both on the number and type of persons entering the plant and how often they enter the plant. These facilities will be large enough to accommodate the normal staff complement as well as contract workers and other persons needed during plant outages. Adequate change room, office, locker, and parking facilities are important to accommodate large outage work crews and to facilitate plant outage operations. Adequate service facilities will improve personnel traffic flow patterns, including those through radiation areas. This improved traffic flow will result in lower collective doses to personnel, and the staff finds this to be acceptable. Service facility provisions for plant security and security training are provided in Chapter 9 of the passive Requirements Document.

4 POWER GENERATION COMPLEX

4.1 Common Requirements

4.1.1 Building and Equipment Arrangement

Section 4.2.2 of Chapter 6 of the passive Requirements Document specifies the general requirements related to the physical location of equipment and the routing of systems within the power generation complex (PGC). Building structures will be used for radiation shielding to the extent practicable. Pipe penetration areas outside the containment will be separated into radioactive and nonradioactive areas. Equipment and spaces required for postaccident operations will be arranged so that they are accessible under postaccident conditions in order to minimize personnel exposure. All radioactive piping will be pre-engineered and not field run. This will ensure that all radioactive piping receives a comprehensive ALARA (as low as is reasonably achievable) design review before installation. Radioactive and nonradioactive equipment and piping will be separated to avoid unnecessary exposure from the radioactive components when the nonradioactive components are being serviced. Radioactive components that are significant sources of radiation exposure will be located in separate cubicles with adequate laydown space for maintenance provided in each cubicle. Radioactive components that require little maintenance (such as filters and tanks) will be separated from radioactive components of the same system that require more maintenance (such as pumps and valves) to minimize radioactive exposure during maintenance of the more passive components. Also, redundant radioactive systems will be separated (shielded) from each other to permit maintenance on one system while the other is in service. Separate sumps will be provided for potentially contaminated drains and clean drains to prevent the spread of contamination. Cubicles containing radioactive components will have access labyrinths of the correct size to permit the removal and replacement of the equipment within the cubicle, yet they will be designed to preclude any radiation streaming from the cubicle into the access corridor or into areas of low background radiation. These requirements would reduce occupational exposure and facilitate maintenance by separating radioactive and nonradioactive components and piping.

The staff concludes that the general requirements of Section 4.2.2 of Chapter 6 represent good design practice, are consistent with the guidelines of SRP Sections 3.4.1, "Flood Protection; and 3.5.1.1, "Internally Generated Missiles (Outside Containment)" and are, therefore, acceptable. In addition, these design features, which are intended to minimize occupational exposures by the separation of radioactive and nonradioactive components and piping, conform with the guidelines of RG 8.8 and are acceptable.

4.1.2 System Supports

Section 4.2.3.2 of Chapter 6 states that supports for all non-ASME seismic Category I piping will be designed in accordance with ANSI/American Institute of Steel Construction (AISC) N-690, "Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities." In a letter dated May 17, 1991, the staff stated that since it had not yet

endorsed this standard for piping supports, the only acceptable rules for non-ASME seismic Category I piping are found in ANSI/ASME B31.1 "Power Piping." Therefore, the staff requested that Section 4.2.3.2 be revised to reflect this position. In a letter dated August 1, 1991, EPRI responded to this request by stating its position that a change to the passive Requirements Document was unnecessary because ANSI/AISC N-690 was developed explicitly for the nuclear industry and should be appropriate for the design of the next generation of nuclear plants. The staff does not agree with EPRI's position. The staff has not endorsed N-690 for use in the design of supports for ASME piping. Further, the staff is not even considering N-690 for use in the design of non-ASME seismic Category I piping. Therefore, at this time, the staff's position remains as stated above, that is, ANSI/ASME B31.1 contains the only acceptable rules for use in the design of supports for non-ASME piping. The staff will review an individual application for FDA/DC in accordance with the position stated above.

The NRC staff is addressing supports for ASME piping by participating in the program to incorporate N-690 into ASME Code, Section III, Subsection NF, "Component Supports." When this program has been completed and endorsed by the staff, Subsection NF will still be the standard to be used for the design of ASME component supports; however, portions of N-690 will be included in that standard. Table 1.4-1 in Chapter 1 of the passive Requirements Document references Subsection NF as the standard to be used in the design of ASME Class 1, 2, and 3 component supports. This is consistent with the staff's position and is acceptable.

Section 4.2.3.3 of Chapter 6 requires that the plant designer prepare drawings to adequately define all support details. For ASME pipe supports, a unique drawing will be developed for each support. For clips or other small supports used in a number of locations, a typical drawing will be used. Critical dimensions and tolerances will be clearly identified. The supports will be located on the system drawings relative to the system hardware or components (i.e., valves and elbows). Reference to column lines will be included for information only. A complete bill of materials will be included on the drawing. The plant designer will specify the inspection attributes. The welding documentation, that is, weld data sheets for all field welds, will be included with the drawing.

These requirements are acceptable because drawings prepared in this manner will simplify the installation. The inclusion of weld data sheets with the drawings will provide the necessary controls to ensure adequate welding quality.

Section 4.2.3.5 of Chapter 6, Revision 1, states that alternate seismic restraint devices, such as energy absorbers, that have been accepted in the ASME Code and by other regulatory agencies, will be used in lieu of snubbers, where practicable. The only alternate devices that the staff is currently accepting without a plant-specific review are those identified in ASME Code Case N-420, "Linear Energy Absorbing Supports for Subsection NF, Classes 1, 2, and 3 Construction, Section III, Division I." Code Case N-420 was conditionally accepted by the staff in RG 1.84, "Design and Fabrication Code Case Acceptability--ASME Section III, Division 1," Revision 24, dated June 1986. In addition, one of the conditions of the staff's approval of Code Case N-411, "Alternative Damping Values for Seismic Analysis of Classes 1, 2, and 3 Piping

Sections, Section III, Division 1," Revision 24 of RG 1.84, is that Code Case N-411 cannot be used in analyses for piping systems in which linear absorbing supports covered by Code Case N-420 are used. This latter issue is discussed in Chapter 1 of this report. The revised requirement in Section 4.2.3.5 of Chapter 6 (as clarified by the staff's position relative to the use of energy absorbers, and the referenced discussion in Chapter 1 of this report) is acceptable.

In addition, the surveillance and maintenance of snubbers can increase occupational exposures as a result of such problems as fluid leakage in hydraulic snubbers and binding problems in mechanical snubbers. The placement of these snubbers in certain parts of the plant can impede access for maintenance and operational purposes. EPRI states that the design and layout of seismic supports will take into account operation and maintenance access requirements. Use of these supports will be minimized in contaminated areas. The proposed restrictions on the use and location of seismic supports are intended to lower the potential for occupational exposures during surveillance and maintenance operations and are, therefore, acceptable.

4.1.3 Piping Arrangement

In Section 4.2.4 of Chapter 6, EPRI specifies the use of the following piping design features to eliminate crud traps and thereby minimize the buildup of crud in radioactive piping:

- location of expansion loops in the horizontal plane
- use of butt-welded connections with long radius bends in resin piping
- connections to horizontal piping runs designed to enter the run vertically or diagonally above its center line
- minimization of low points, drains, vents, and other configurations that collect crud
- use of butt welds without backing rings in all radioactive system piping more than 2 inches in diameter
- sloping of drain and sampling lines not continuously used during normal operations to prevent crud pockets or areas of stagnant fluids
- orientation of valve stems so that crud does not settle in the bonnets
- flow-restricting devices designed to minimize the accumulation of crud

These piping design features conform with the guidelines of RG 8.8 for reducing crud and are acceptable.

EPRI states that each valve or pump that handles radioactive fluid and is not self-draining will have a designated drain path to a floor drain. All floor drains from potentially contaminated areas will be shielded if they are routed through a clean area that will be normally or periodically occupied. In addition, EPRI specifies that floor drains will have adequate traps to prevent radioactive gas from moving from compartment to compartment through the drain

lines. All valves will be readily accessible for maintenance. Those valves located in high-radiation areas will be operated, where practicable, by controllers located outside the high-radiation areas. These features are intended to reduce occupational exposure during valve maintenance and operation and are acceptable.

In Table B.1-2 of Appendix B to Chapter 1, EPRI commits to comply with RG 1.151, "Instrument Sensing Lines." However, the requirement in Section 4.2.4.33 of Chapter 6, Revision 2, does not agree with the guidelines of RG 1.151. The requirement states that instrument-sensing lines connected to ASME fluid systems will be designed in accordance with ASME Code, Section III, and supports for these lines must be designed in accordance with ANSI/AISC N-690. This requirement should state that all safety-related instrument-sensing lines will be classified as applicable in accordance with the guidelines in RG 1.151. Those lines classified as ASME Class 2 or 3 will be constructed in accordance with ASME Code, Section III, Subsection NC or ND, respectively. The only permitted exception to these rules is for supports for these lines. Supports will be designed, but not necessarily constructed, to the rules in ASME Code, Section III, Subsection NF. Supports will be seismic Category I and the pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 will be applied to all activities affecting the safety-related functions of these supports. As stated above, the staff has not yet endorsed ANSI/AISC N-690 for the design of piping supports but will evaluate its use during an individual application for FDA/DC. Until then, the staff will review an individual application for FDA/DC in accordance with the above position.

Section 4.2.4.27 of Chapter 6 requires that the containment isolation valves be located as close to the containment as practical. Sufficient space will be provided between the valves and containment boundary to permit the following: (1) inservice inspection (ISI) of nonisolable welds, (2) Appendix J to 10 CFR Part 50 leak testing, (3) cutout and replacement of the isolation valves using standard pipe fitting tools and equipment, (4) local control, and (5) in-place resurfacing of valve seats. These requirements are acceptable because sufficient space will be provided to permit ISI of nonisolable welds.

4.1.4 Heating, Ventilation, and Air Conditioning

To prevent the spread of contamination, the heating, ventilation, and air conditioning (HVAC) system will route air from cleaner areas to contaminated or potentially contaminated areas. HVAC duct penetrations through rooms and cubicles containing potentially high-radiation sources will be located above head level so that personnel outside the rooms or cubicles will not be exposed to radiation streaming from inside the rooms or cubicles. These HVAC design features are consistent with the guidelines of RG 8.8 and are, therefore, acceptable.

4.1.5 Inservice Inspection Considerations

Section 4.2.7 of Chapter 6 specifies the general access requirements of building design and arrangement to facilitate ISIs performed in radiation areas and thereby minimize occupational radiation doses. This section mandates the following requirements for ISI:

- The functional requirements of ISI will be as defined in Section XI of the ASME Code, supplemented by the NRC guidelines as given in the SRP (NUREG-0800).

- The arrangement designs of the power generation buildings will allow for ISI of designated piping, vessels, and components.

- Piping and pipe support locations, insulation, hangers, and stops will be designed so as not to interfere with inspection equipment and personnel. Where this cannot be done, the components must be easily and quickly removable with minimal special handling equipment.

- Permanent access platforms will be provided outside reactor pressure vessel shield walls to facilitate ISI of the reactor vessel and nozzles. This will include access to the vessel bottom or bottom nozzles.

- Radial clearances of at least 6 inches will be provided around pipe or component welds requiring volumetric or surface examination for ISI.

- Mechanized inspections will be considered for areas in which radiation levels exceed 50 mrem/hr or physical limitations restrict or prevent manual methods.

- Piping arrangement will allow for adequate separation of piping welds so that space will be available to perform ISI. Component nozzles, tees, valves, elbows, branch connections, support lugs, and other fittings will be separated by sections of piping with a length of at least three times the wall thickness or 6 inches from weld to weld, whichever is longer. Welds in piping that passes through walls will be located away from the wall by at least 6 inches or three times the pipe wall thickness, whichever is longer.

- ISI welds requiring volumetric examination will be located in areas with adequate access for personnel and ultrasonic testing equipment. Platforms and walkways will be provided as necessary for access to the welds requiring volumetric or surface examinations.

- Where in-place testing of safety-related valves is a requirement, provisions will be made for access.

- ISI weld profile and finish requirements will be clearly identified on the weld data sheets. Weld crown profile and surface finish will be specified suitable for future surface or volumetric ISI for the weld.

- Reliable position indicators will be provided on safety-related active check valves.

These requirements are acceptable because they ensure that nuclear power plant components and component welds will be accessible for ISI.

The access requirements of Section 4.2.7 of Chapter 6 for ISI are in accordance with the recommendations of the Atomic Industrial Forum, Inc. (AIF) study reported in AIF/NESP-020, "Compendium of Design Features To Reduce

Occupational Radiation Exposure at Nuclear Power Plants." These recommendations comply with the guidelines of RG 8.8; represent desirable design practice; and are acceptable with respect to ALARA considerations.

4.1.6 Radiation Zones and Shielding

Section 4.2.8 of Chapter 6 specifies the general design requirements related to radiation zones and shielding. EPRI specifies that radioactive equipment will be located in shielded compartments and that shielding will be placed between radioactive and nonradioactive components and between redundant radioactive components located in the same cubicle. Removable local shielding will be provided for equipment that could exceed the radiation zone designation. Space, support points, and handling facilities will be provided for the placement of this temporary shielding. Local control panels will be located in low-radiation areas. Buildings will be arranged so as to make maximum use of common walls and floors for shielding. EPRI states that radioactive piping running through normally accessible areas will be routed through shielded pipe chases to minimize occupational exposure to personnel.

These requirements of Section 4.2.8 represent good design practice and are acceptable. Because the staff is concerned about the potentially excessive number of radiation overexposure events involving reactor cavities and fuel transfer tubes that have occurred in operating reactors, it will, in its reviews of applications for FDA/DC, pay particular attention to access controls and shielding for these areas.

The radiation zones established, along with the radiation source terms, will serve as the basis for the access control measures, building layout, equipment design and layout to maintain assigned radiation levels, and the calculation of radiation shielding thicknesses. The radiation zone designations and radiation zone maps will be provided by the plant designer and will be consistent with the guidelines of RG 8.8.

The staff concludes that the shielding design features conform with the guidelines of RG 8.8 and are acceptable.

4.1.7 Design of High-Radiation Areas

Section 4.2.9 of Chapter 6 specifies the general requirements related to the design of high-radiation areas. EPRI specifies that ALWRs will be designed to minimize or eliminate the need for personnel to enter areas where the dose rate will be greater than 100 mrem/hr during routine operation.

The following design features proposed by EPRI are intended to minimize radiation levels where personnel will be required to perform operational and maintenance functions:

- Remote viewing devices will be provided for routine visual surveillance.
- Valve operators, instrumentation indicators, instrument isolation valves, and sensing element transmitter readouts will be located in low-radiation zones.

- Instrumentation will be designed for long service life and low frequency of maintenance and calibration.

- Reach rod shafts for valves will not block or cross corridors that must be accessed for maintenance of equipment.

- Remotely controlled TV cameras with scanning and zoom capabilities will be used for remote surveillance of high-radiation areas.

- Valves will be located and shielded to enable valve maintenance without significant exposure from nearby radioactive components.

A number of overexposures or potential overexposure events have occurred at operating reactors because of operational occurrences (such as spent fuel transfer and irradiated traversing in-core probe or thimble tube movement) which have resulted in transient high-radiation levels. EPRI requires that plant designers evaluate all areas in which transient high radiation could be experienced and design these areas to minimize the potential for large radiation doses (greater than 100 mrem/hr) from such transient sources. The staff addresses this issue further in Chapter 7 of this report. During its review of an individual application for FDA/DC, the staff will ensure that the applicant has identified all potential high-radiation areas. The staff will also review the applicant's measures to minimize inadvertent personnel exposure from these areas.

The design features of Section 4.2.9 of Chapter 6 are based on design features described in RG 8.8, will facilitate compliance with ALARA requirements, and are, therefore, acceptable.

4.1.8 Contamination Control

Section 4.2.10 of Chapter 6 specifies the general requirements for floor surface sloping, surface coatings, and tank curbs to reduce the spread of contamination and to facilitate cleanup.

Cubicles will be designed to reduce the potential for the spread of radioactive contamination and to facilitate cleanup. Floor surfaces will sloped to drains that will be of the proper size to handle cleanup water flow rates. Tanks containing contaminated liquids will have curbs to contain any potential leaks or spills. Floors and walls of cubicles that may become contaminated from radioactive leaks or spills will be protected with a smooth-surface epoxy coating to facilitate decontamination.

The requirements of Section 4.2.10 of Chapter 6 represent good design practice for minimizing the spread and facilitating the cleanup of radioactive contamination, conform with the guidelines of RG 8.8 for radiation protection, and are, therefore, acceptable.

4.1.9 Structural Design

Section 4.2.11 of Chapter 6 specifies the structural design requirements for (1) mats and foundations, (2) interaction between structures, (3) building frames, (4) building layout, (5) concrete design, (6) reinforcing steel

design, (7) structural steel design, and (8) platforms, ladders, and stairways. These requirements are consistent with the applicable American Concrete Institute (ACI) and AISC codes and standards and current regulatory guidance in SRP Sections 3.7 and 3.8, and are, therefore, acceptable. The staff, however, was concerned about the design requirements for both mat foundation and rebars in water-retaining concrete structures. In its letter dated April 24, 1991, requesting additional information, the staff asked EPRI to pay careful attention to the following items: (1) construction joints and water stops, (2) construction sequence and associated loading, (3) asymmetrical loadings on the mat, (4) potential of differential settlement between lightly loaded and heavily loaded portions of the mat, (5) effect of varying concrete thickness with respect to potential for inferior concrete placements at the transitions resulting from construction difficulties, and (6) potential complexity for soil-structure interaction analysis and structural modeling parameters.

In its response dated November 18, 1991, EPRI acknowledged the importance of such considerations in the design process. However, EPRI believes that these considerations are similar to those necessary for proper design of independently founded structures. EPRI noted that construction joints and construction equipment loadings are addressed by ACI 349 and these considerations can be just as significant for individual buildings as for integrated structures on a common basemat. Regarding the potential for differential settlement, EPRI stated that the use of a common basemat actually enhances the performance of structures because the basemat and principal walls and slabs above the basemat will provide continuity across designated building boundaries; hence, the basemat will be stiffened by the buildings that tend to act as one integrated unit. Shear load transfer from heavily loaded portions of the building complex to more lightly loaded areas would be more difficult, but the interconnecting shear wall matrix will facilitate this load transfer. The staff concludes that EPRI's explanation of load transfer phenomena appears reasonable; however, the staff will evaluate the design of the common basemat during its review of an individual application for FDA/DC.

In its letter dated May 5, 1993, EPRI revised Section 4.2.11.6.3 of Chapter 6 regarding the requirements for splicing of reinforcing steel. The requirements in Section 4.2.11.6.3 of Chapter 6 specify that in areas where reinforcing steel splicing is necessary and lap splices are precluded, splicing will be in accordance with the applicable code requirements. The alternatives for qualified field splicing will include thread deformed, taper thread, swaged, and ferrous metal filler types. The staff concludes that these requirements for mechanical splicing of reinforcing steel are adequate and acceptable.

In its letter dated April 24, 1991, the staff requested that Section 4.2.11.6 of Chapter 6 include guidance related to rebar design for water-retaining concrete structures (such as extra rebar clearances and use of epoxy-coated bars). In Revision 3, EPRI added Section 4.2.11.6.4 to Chapter 6, requiring the plant designer to evaluate the potential for corrosion of reinforcing bars at intake structures exposed to salt or brackish water and to provide for epoxy-coated rebars in such highly corrosive environments. The use of epoxy-coated rebar for preventing corrosion is acceptable. In the DSER for Chapter 6, the staff recommended that EPRI reconsider the criteria (or requirement) for calculating the development length because the epoxy coating will

reduce the bonding between rebar and concrete and identified this as a vendor- or utility-specific item. EPRI has revised Section 4.2.11.6.4 to require that the development length of epoxy-coated rebar for safety-related structures other than the containment be calculated by formulas specified in Section 12.2.2 of ACI 349 and multiplied by factors specified in Section 12.2.4.3 of ACI 318-1989. For non-safety related structures, the requirements of ACI 318-1989 must be followed. The criteria for development length of epoxy-coated rebar that are provided in ACI 318-1989 are based on tests performed at the University of Texas. The staff concludes that these criteria provide a reasonable method to account for reduction in the bonding between the rebar and concrete when epoxy-coated rebar is used. This issue is considered closed.

Section 9.4.1 of Chapter 11 provides the requirements to determine the need for cathodic protection. The rationale portion of this section states that in view of the extended plant life, it is expected that metal surfaces in contact with soil, such as tanks and buried piping, will be provided with cathodic protection. The staff concludes that these requirements adequately address the corrosion protection requirements for buried structures.

During plant outages, the setup and breakdown of temporary scaffolding in radiological control areas can be a significant contributor to outage dose. EPRI specifies that permanent platforms will be provided in areas of multiple construction and inspection operations instead of multiple temporary scaffolding setups. This will result in lower occupational exposure as well as less congestion in radiological control areas during outages. To facilitate access, especially in high-radiation areas, stairways are preferable to ladders. The staff concludes that these features are intended to lower occupational exposure and are, therefore, acceptable.

4.1.10 Crane Path Routing

Section 4.2.13 of Chapter 6 requires that safe load paths be developed for moving all heavy or critical loads above and across the areas served by all cranes in the power generation complex.

This requirement is consistent with the requirements of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," and is, therefore, acceptable.

4.2 Primary Containment Structure

4.2.1 General Requirements

Section 4.3.2 of Chapter 6 of the passive Requirements Document delineates the general requirements for primary containment structures. Section 4.4 and Table 1.4-3 of Chapter 1 of the passive Requirements Document states that the latest version of the ASME Code, Section III, Division 2, "Code for Concrete Reactor Vessels and Containments," will be used for the design of primary containment structures. Section 4.3.2.4 of Chapter 6, also states that for a reinforced-concrete containment, advantage must be taken in the design process of the relaxation of code requirements for structural rebars that EPRI anticipates will take place. For example, EPRI anticipates that current code requirements for diagonal rebars in reinforced-concrete containments are likely to be eliminated. The staff has reviewed the adequacy of the ASME

Code, Section III, Division 2 for reinforced-concrete containment design and concludes that it is acceptable, with the following clarification of the criteria for tangential shear design. According to Regulatory Guide 1.136 (Rev. 2), the staff has not yet endorsed Subsection CC-3000 of the ASME Code, Section III, Division 2, "Code for Concrete Reactor Vessels and Containments." Specifically, the staff raised a concern that the code-specified allowable tangential shear stress of 0.2 f_c' (without installation of diagonal rebars) is too high for reinforced-concrete containments. It is the staff's position that if the diagonal rebars are not used, the allowable tangential shear stress should be limited to 2.65 $\sqrt{f_c'}$ as specified in ACI 318 for the inplane shear and the total shear strain should be limited to 0.0003. The staff will evaluate this issue during its review of individual applications for FDA/DC.

Section 4.3.2.5 of Chapter 6 states that where hydrogen mixing and dilution will be used as a means of hydrogen control, the containment will be arranged so as to maximize natural circulation mixing and to prevent local hydrogen accumulation. Small, enclosed spaces with a source of hydrogen will be avoided. Geometries that tend to permit flame acceleration and deflagration-to-detonation transition will be minimized. Features such as vents and floor gratings instead of solid flooring will be used where possible to promote mixing. The staff addresses hydrogen control in Chapter 5 of this report.

Section 4.3.2.6 of Chapter 6 states that provisions will be available for flooding PWR reactor cavities and BWR drywells with water to cool core debris in the event of a severe accident involving reactor vessel melt-through. EPRI states that passive means are preferred to flooding and that automatic systems should not be used. The source of water will be the in-containment refueling water storage tank (IRWST) for PWRs and may involve the suppression pool for BWRs. A backup source of water from outside the containment will be provided. Section 4.3.2.6 requires that the cavity/drywell arrangement preclude direct contact between core debris and the containment boundary. A minimum 3-foot-thick concrete barrier will protect the steel containment or the containment liner from core debris.

The design of the reactor vessel cavity/drywell is discussed by the staff in Chapter 5 of this report.

Section 4.3.2.7 of Chapter 6 requires that paints and coatings that will be exposed to containment atmosphere be qualified in accordance with ANSI N101.2-1972 (including the quality assurance requirements) and American Society for Testing and Materials (ASTM) D3842-80.

These requirements are acceptable because the paints and coatings will be properly qualified to perform as designed. However, in the DSER for Chapter 6, the staff identified a typographical error in regard to the ANSI reference. In Revision 3, EPRI revised the section to reference ANSI N101.4-1972 instead of ANSI N101.2-1972.

Section 4.3.2.12 of Chapter 6 requires that the wetted surfaces of all passive safety system tanks that will contain demineralized water during normal operation of the plant, where the water could be used in the reactor coolant system or for refueling purposes, be constructed of or lined with low carbon austenitic stainless steel material, such as Type 304L. The material used for the tank supports will be compatible with the tank material.

This requirement is acceptable because these materials will provide adequate corrosion resistance against demineralized water for all passive safety system tanks.

4.2.2 BWR Primary Containment Structure

To facilitate maintenance of reactor internal pumps, control rod drives, and incore instrumentation, Section 4.3.3.7 of Chapter 6 specifies that an equipment platform and hoisting facilities will be located under the reactor pressure vessel in the lower drywell. Specific tools and other equipment regularly used to perform this maintenance will also be located in the drywell to improve the effectiveness of working in the drywell. EPRI states that these tools will be designed to require minimal maintenance and to have a long service life. These features are intended to minimize unnecessary traffic in the drywell during outages and thereby reduce personnel exposures. The staff finds this requirement acceptable.

In its July 3, 1989, response to the staff's RAI, EPRI stated that the shielding and arrangements of the advanced BWR design will be such that movement of irradiated fuel from the reactor vessel to the fuel pool will not restrict access to the drywell for maintenance or inspection activities. Irradiated fuel transfer from the reactor vessel into fuel storage could result in potentially lethal radiation levels in the upper levels of the drywell if, during such transfer, the spent fuel assembly was dropped while being moved over the reactor vessel/fuel pool boundary area. Other fuel rod configurations during fuel transfer could also result in very high radiation levels in portions of the drywell. The applicant for design certification should describe those design features of the advanced BWR that will preclude the occurrence of potentially lethal radiation levels in portions of the drywell during the transfer of irradiated fuel from the reactor vessel into fuel storage. This issue will be evaluated by the staff during its review of an individual application for FDA/DC.

Section 4.3.3.8.2 of Chapter 6 requires that the wetted surfaces of the wetwell and the vent system be Type 304L stainless steel. The plant designer will determine the application of stainless steel cladding or solid stainless steel plate, depending on cost effectiveness. If stainless steel cladding is used, the minimum thickness will be 1/8 inch. This requirement is acceptable because Type 304L stainless steel material will provide adequate corrosion resistance to the BWR primary containment.

To ensure applicants meet the requirements of 10 CFR 73.55(d)(8) for reducing the threat of sabotage within the primary containment, the staff will evaluate access to the reactor containment, including shutdown access, during its review of an individual application for FDA/DC.

4.2.3 PWR Primary Containment Structure

Section 4.3.4.1.1 of Chapter 6 specifies that the plant designer will establish and justify a corrosion allowance for the steel containment vessel. Regions of the steel shell that are not fully embedded in concrete, on both interior and exterior surfaces, will be designed to permit inspection for the purpose of determining the integrity of the steel shell. Inspection capability using fiber optics or other remote visual technique will be permitted.

These requirements are acceptable because they provide for inspecting interior and exterior containment surfaces to detect degradation.

One of the more dose-intensive jobs at PWRs involves the replacement of steam generators. Section 4.3.4.3 of Chapter 6 states that the containment arrangement will permit the removal and replacement of all steam generators during a 100-day plant shutdown. The size of the maintenance hatch will permit the removal and replacement of a reactor coolant pump and will be located in the same containment building quadrant and on the same level as the health physics access control area. This will facilitate access to and egress from the containment during outages (thereby minimizing unproductive time and effort) and will improve control of the spread of radioactive contamination.

Section 4.3.4.4.5 of Chapter 6 requires that the in-containment refueling water storage tank (IRWST) be arranged to facilitate ISI of the high-pressure, high-temperature, passive decay heat removal (PDHR) heat exchangers located within the IRWST. This requirement is acceptable since it allows ISI to be performed as required by Section XI of the ASME Code.

Section 4.3.4.5 of Chapter 6 states that the reactor vessel cavity (PWR) will be arranged to prevent transport of significant quantities of core particulate debris into the large open portion of the containment or into direct contact with the containment boundary after a postulated vessel failure. Analyses using the Modular Accident Analysis Program will be performed to confirm that the cavity pathway, cavity exit, and core debris de-entrainment capability are adequate to confine debris to the cavity and lower containment volume and avoid early containment failure from direct containment heating.

EPRI describes a configuration for PWRs that would be expected to provide features that will limit direct containment heating, including:

- a cavity exit area that is significantly greater than the area around the vessel at the refueling seal
- a cavity pathway from the reactor vessel to the collection volume that is horizontal and that provides a preferential flow path for core debris to the collection volume
- a collection volume of at least twice the total core material volume
- a cavity exit that is perpendicular to the cavity and is located far enough from the reactor vessel to prevent streaming of core debris directly into the upper containment volume

Additionally, where junction boxes of electrical penetrations terminate within the annular space between the containment structure and the shield building, a secondary enclosure will be included to serve as an additional barrier to potential leakage at the electrical penetrations.

Although the conceptual general arrangement as described in the passive Requirements Document for reactor cavities would provide retentive cavity features, aerosols and radioactive gases (which could not be trapped as easily by geometric configuration) would also contribute to the offsite dose. The features discussed would be consistent with current design concepts which

appear to have retentive features. As for the retention of aerosols and gases, the staff's evaluation of the retention and plateout of fission products within the primary containment following an accident is provided in Section 2.1.6 of Chapter 5 of this report.

Section 4.3.4.6 of Chapter 6 states that the PWR containment building arrangement will ensure that all compartments that could receive discharge from the primary system in a severe accident before vessel failure will discharge directly into the main natural circulation flow loop within the containment for improved mixing capability. Compartments that could receive primary system discharge before vessel failure will be arranged to discharge directly into the natural circulation flow path.

Section 4.3.4.7 of Chapter 6 states that three access penetrations will be provided: (1) a maintenance hatch located at grade level, (2) a personnel air lock located at the operating deck level, and (3) a personnel air lock located at grade level.

To ensure applicants meet the requirements of 10 CFR 73.55(d)(8) for reducing the threat of sabotage within the containment, the staff will evaluate access to the reactor containment, including shutdown access, during its review of an individual application for FDA/DC.

Section 4.3.4.8 of Chapter 6 states that the designer will provide a polar crane that complies with all the requirements of ANSI/ASME NOG-1-1983, "Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)," for Type II cranes. In addition to lifting equipment or components during refueling operations, the crane will be capable of lifting a steam generator clear of its shield wall without having to remove sections of the shield wall. The electrical components of the crane, including controllers, motors, sensors, and controls, will be qualified for the design operating temperature, radiation, humidity, and mist in the containment dome and will be designed in modular plug-in form to facilitate the replacement of faulty components. Section 4.3.4.9 of Chapter 6 states that a minimum of two small (10-ton) jib cranes will be provided on the operating deck, located so as to provide lifting capabilities to all major vertical access ways and adjacent work areas. In addition, dedicated cranes will be provided for the maintenance hatch and reactor head service area. ANSI/ASME NOG-1-1983 has not been endorsed by the staff. As discussed in Section 4.4 of Chapter 1 of this report, instances where the plant designer uses codes and standards that conflict with NRC positions must be resolved with the NRC staff and fully documented during the review of an individual application for FDA/DC.

Section 4.3.4.12 of Chapter 6 specifies the following requirements for the passive containment cooling system (PCCS):

- A PCCS will be provided to serve as the ultimate heat sink for removal of reactor coolant system heat and core decay heat from the containment building following a design-basis event or if the containment coolers are not available. This system will also provide long-term containment cooling following a severe accident.

- The PCCS will consist of an elevated tank of water and the necessary piping to provide cooling water flow over the outside surface of the

steel containment structure. The PCCS storage tank will hold enough water to be able to cool the containment surface for 3 days.

- Material for the wetted surfaces of the PCCS tank will be selected on the basis of a cost comparison between coated carbon steel, low-carbon austenitic steel, or such materials as fiberglass.

The PCCS is reviewed by the staff in Chapter 5 of this report.

4.3 BWR Reactor Building and PWR Auxiliary Building

4.3.1 BWR Reactor Building

Section 4.4.2 of Chapter 6 of the passive Requirements Document states that the BWR reactor building will contain the engineered safety systems, the reactor, the reactor coolant system, and non-safety-grade auxiliary systems. This building will be located within the plant protected area. The staff considers that locating the redundant non-safety-grade auxiliary systems within the plant protected area may be important in providing defense-in-depth protection against sabotage.

To reduce personnel radiation exposure, "clean" and "controlled" access areas will be segregated and personnel will preferably enter the reactor building on the same level as the radioactive work change areas. Rooms will be provided in the reactor building so that safety/relief valves and main steam isolation valves can be serviced within the controlled zone without having to completely decontaminate them and move them out of the reactor building for servicing. These features are intended to minimize personnel radiation exposure and control the spread of contamination and are, therefore, acceptable.

4.3.2 PWR Auxiliary Building

Section 4.4.3 of Chapter 6 states that the PWR auxiliary building will contain the non-safety-grade auxiliary and passive safety systems that support the primary reactor systems. This building will be located within the protected area. The staff considers that locating the redundant non-safety-grade auxiliary systems within the protected area may provide important defense-in-depth protection against sabotage.

As discussed in Section 4.4.2 of Chapter 6, clean and controlled areas of the PWR auxiliary building will be segregated and personnel will enter the building on the level on which the radioactive work change areas are located. These features are intended to minimize personal radiation exposure and control the spread of contamination. The staff, therefore, finds them to be acceptable.

4.4 Turbine-Generator Building

In letters dated May 17 and August 29, 1991, the staff submitted RAIs in which it questioned how the BWR main steamline beyond the seismic restraint up to and including the turbine stop valve, the turbine bypass lines to the condenser, and the condenser will be protected from the possible failure of non-seismic Category I structures, components, and systems during a postulated safe shutdown earthquake (SSE). EPRI responded to these requests in a letter

dated November 6, 1991. The staff's evaluation of the information in these responses concerning the BWR turbine-generator building is given in Section 3.1.1 of Chapter 13 of this report.

Section 4.5.2 of Chapter 6 of the passive Requirements Document gives the general design requirements for the turbine-generator building. Section 4.5.2.1 classifies the turbine-generator building as non-seismic Category I, in accordance with the definition in RG 1.29, "Seismic Design Classification," and requires that the building be designed, at a minimum, to the provisions of the Uniform Building Code (UBC). In the DSER for Chapter 6, the staff stated that this seismic classification was not consistent with the classification defined in Section 4.3.2 of Chapter 1 of the passive Requirements Document. According to Section 4.3.2 of Chapter 1, the turbine building would be classified as either seismic Category II or non-seismic depending on whether or not the failure of the building structure during and after the SSE could impair the safety functions of any seismic Category I items. The use of UBC Zone 2A without justification for the turbine building design is not acceptable to the staff. This was identified as a open issue in the DSER.

In a letter dated September 28, 1992, EPRI responded to this issue by reclassifying the turbine building as a non-seismic Category I structure and requiring this building be designed for the UBC Zone 2A requirements with an importance factor of 1.25. EPRI also committed to require the plant designer to demonstrate that the turbine building would not collapse under an SSE assuming structural behavior within acceptable ductility limits. The staff's position is that, at least for the BWRs, the turbine building should be designed, as a minimum, to the provisions of UBC Zone 3 with a restriction that the dual systems and concentric braced frames shall not be used. As an alternative, the provisions of UBC Zone 2A may also be used for the turbine building design, provided the plant designer demonstrates that the turbine building structure will retain its structural integrity under the specified SSE. The staff will review an individual application for FDA/DC to determine the acceptability of the plant designer's demonstration of the structural integrity of the turbine building under the specified SSE and assurance of no adverse interaction with seismic Category I structures, systems, and components. Therefore, this DSER open issue is closed.

EPRI states that the effects of hydrogen injection on N-16 levels will be considered in the design of the BWR turbine-generator building. The turbine-generator building will contain sufficient laydown space for maintenance of the turbine-generator and areas for decontaminating equipment and for cleaning filters. These features, which are intended to maintain occupational exposure levels ALARA, are acceptable.

4.5 Other Power Generation Complex Facilities

4.5.1 Fuel Handling and Storage Facility

Section 4.6.2 of Chapter 6 of the passive Requirements Document gives the design criteria for the fuel handling and storage facility. Section 4.6.2.3 originally required that the facility be designed according to the general structural and loading requirements specified in Chapter 1, "Subsection 1.4" of the passive Requirements Document. In the DSER for Chapter 6, the staff stated that although this requirement was acceptable as discussed in the DSER

for Chapter 1, Section 4, the staff believed that "Subsection 1.4" was a typographical error and the reference should have been "Section 4."

4.5.2 Radwaste Building

Section 4.6.3.3 of Chapter 6 requires that the radwaste building design and equipment supports meet the guidelines of RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," and that the building design comply with the standards of ACI 318 and AISC. This requirement is acceptable. EPRI states that the radwaste facility will be designed so that the waste transport truck will not be contaminated and the expected doses to truck drivers will be comparable to background levels. Radwaste piping carrying fluids with high solids content will be designed to minimize the potential for plugging by the use of short runs and long radius bends. Automatic flushing after each use will be provided for this piping.

To maintain doses to radwaste operators ALARA, EPRI requires that special radwaste-handling methods, including robotics, local lifting equipment, shielding bells, and automatic equipment, be used in the radwaste facility. In addition, all access halls and the radwaste control station will be located in low-radiation zones. These design features for the radwaste facility comply with the guidelines of RG 8.8 for maintaining occupational exposures ALARA and are acceptable.

4.5.3 Emergency Onsite Power Supply Facility

Section 4.6.4 of Chapter 6 provides building design and arrangement requirements for the onsite power supplies required by Chapter 11 of the passive Requirements Document. Each onsite ac generator, which for passive ALWRs is non-safety-grade equipment, will be located in a seismic Category II structure within the protected area, in a compartment designed to prevent sabotage, fire, or explosion in one compartment from interfering with operation of the other generators. The staff considers that it is important to locate this redundant power supply within the plant protected area in order to provide defense-in-depth protection against sabotage.

4.5.4 Control Complex

The control complex includes the main control room (MCR), the technical support center, auxiliary control rooms, relay rooms, offices, and mechanical support equipment areas. Terminology (definitions) are stated to be consistent with NUREG-0800. Man-machine interface systems are described in Chapter 10 of the passive Requirements Document and are evaluated in Chapter 10 of this report.

In the DSER for Chapter 6, the staff stated that EPRI was still evaluating the location of the control complex and that the staff would evaluate the location when it was provided by EPRI. This was identified as an open issue in the DSER. This position was based on the information in the evolutionary Requirements Document, not on the information in the passive Requirements Document, which requires that the general location of the control complex be between the reactor building and the turbine-generator building. The passive Requirements Document also states that the control complex may be located in a separate

building or may be part of another seismic Category I building. The staff concludes that these provisions for the location of the control complex are acceptable; therefore, this open issue is closed.

Section 4.6.5.4.1 of Chapter 6 requires that the MCR be designed so that it can be maintained as the primary location from which technical personnel can safely operate in case of an accident. For example, control room doors, cable penetrations, and other penetrations of the MCR boundary will be designed to minimize leakage into and out of the MCR. The design will preclude air, steam, smoke, water, and firefighting chemicals (such as carbon dioxide) from leaking into the MCR. The MCR ventilation equipment air intakes will be located and protected to minimize the potential for the intake of other than fresh air. EPRI's requirements regarding control room habitability are discussed in Chapter 5 of the passive Requirements Document and the staff's evaluation is in Chapter 5 of this report. The staff's review of the control complex HVAC system is discussed in Chapter 9 of this report. The staff will evaluate the possible need for two remote HVAC air intakes in order to meet the dose criteria of General Design Criterion (GDC) 19 in Appendix A to 10 CFR Part 50, during its review of an individual application for FDA/DC.

Section 4.6.5.4.2 of Chapter 6 states that a control room emergency zone encompassing the control room, critical documents file, computer room, shift supervisor's office, lavatory, kitchen and eating area, and other areas requiring continuous or frequent occupancy under accident conditions will be provided in ALWR designs. EPRI states that the zone will be designed in accordance with human factors guidelines to facilitate operability and to maintain habitability during upset conditions. In the DSER for Chapter 6, the staff stated that in a letter dated August 30, 1990, it had asked EPRI to provide additional information on the requirements and acceptance criteria for human factors considerations to ensure that operability and maintainability were achieved in building arrangements. The staff also requested that EPRI provide information on the scope and objectives of EPRI NP-4350, "Human Engineering Guidelines for Maintainability," which was referenced in Section 2.2.1 of Chapter 6 of the evolutionary Requirements Document. In the DSER for Chapter 6, the staff identified the need for additional information on the requirements and acceptance criteria for human factors considerations and on the scope and objectives of EPRI NP-4350 as an open issue. The staff's evaluation of human factors considerations is given in Chapter 10 of this report. Therefore, this DSER open issue is closed.

The staff concludes that the requirements of Section 4.6.5 of Chapter 6 of the passive Requirements Document do not conflict with current regulatory guidelines and are acceptable.

4.5.5 Technical Support Center

Section 4.6.6 of Chapter 6 states that the technical support center (TSC) will be located so that it is convenient to the control room. EPRI states that the TSC will comply with NUREG-0696, "Functional Criteria for Emergency Response Facilities." NUREG-0696 states that the TSC is to have the same radiological habitability as the control room under accident conditions. The TSC ventilation system need not be seismic Category I qualified or redundant; however, high-efficiency particulate air and charcoal filters are needed. NUREG-0696 also specifies that the TSC is to be provided sufficient alternative or backup

power sources to maintain continuity of TSC functions and to immediately resume acquisition, storage, and display of TSC data, if loss of the primary TSC power sources occurs.

The staff concludes that the EPRI commitment to meet NUREG-0696 is acceptable; however, insufficient information is provided to allow the staff to determine if a design complying with EPRI requirements will meet NRC regulations. Therefore, the staff will evaluate the design of the TSC against the criteria in NUREG-0696 during its review of an individual application for FDA/DC.

5 CONCLUSIONS

The staff concludes that the requirements established in Chapter 6 of the passive Requirements Document for the building design and arrangement do not conflict with current regulatory guidelines and are acceptable. However, by themselves, they do not provide sufficient information for the NRC staff to determine if the plant-specific design and arrangement of the buildings and structures will be adequate. Applicants referencing the passive Requirements Document will be required to demonstrate compliance with the additional guidance in the SRP, or provide justification for alternative means of implementing the associated regulatory requirements.

Therefore, the staff concludes that Chapter 6 specifies requirements that, if properly translated into a design in accordance with the NRC regulations in force at the time the design is submitted, should result in a nuclear power plant whose buildings and structures will perform as designed and have all the attributes required by the regulations to ensure that there is no undue risk to the health and safety of the public or to the environment.

APPENDIX A
DEFINITIONS AND ACRONYMS

Appendix A of Chapter 6 of the passive Requirements Document contains definitions of terms and acronyms. The staff has provided a consolidated list of acronyms in Volume 1 of this report.

APPENDIX B
BWR CONSTRUCTION PLAN AND ARRANGEMENT SKETCHES

Appendix B of Chapter 6 of the passive Requirements Document provides preliminary plant layout sketches for a passive BWR plant. EPRI states that the purpose of the sketches is to delineate the scheduled arrangement characteristics that are to be treated as requirements by plant designers. The sketches are not intended to delineate all of the detailed arrangement features that are requirements of the text, particularly those that would necessitate a detailed level of engineering study and/or specific component sizing to establish final configuration.

APPENDIX C
PWR CONSTRUCTION PLAN AND ARRANGEMENT SKETCHES

Appendix C of Chapter 6 of the passive Requirements Document provides preliminary plant layout sketches for a passive PWR plant. EPRI states that the purpose of the sketches is to delineate the selected arrangement characteristics that are to be treated as requirements by plant designers. The sketches are not intended to delineate all of the detailed arrangement features that are requirements of the text, particularly those that would necessitate a detailed level of engineering study and/or specific component sizing to establish final configuration.

APPENDIX D
MAINTAINABILITY EVALUATION

Appendix D of Chapter 6 of the passive Requirements Document provides a sample format for the report of an evaluation demonstrating that maintenance access and planning requirements of the passive Requirements Document will be met. The purpose of the evaluation; which will be performed by the plant designer, is to ensure the following:

- Refueling can be completed in 17 days.
- Critical maintenance tasks can be performed easily and quickly.
- Satisfactory services can be provided at the access control facility.
- Access to all components is available for their removal and replacement, and these components can be transported using safe, preplanned lifting and transport methods.

Included in this appendix are (1) a list of critical maintenance tasks to be evaluated, (2) a sample maintenance task checklist, and (3) a sample transport task checklist. The evaluations described in this appendix are intended to demonstrate that the completed standard plant arrangement under review will facilitate performing the tasks identified in Section 8 of Chapter 1 of the passive Requirements Document. The tasks referenced in Section 8 of Chapter 1 are those that have historically contributed to extended outages, large percentages of total plant person-rem exposure, and excessive efforts to improve maintainability through plant modifications.

Although the information in Appendix D to Chapter 6 of the passive Requirements Document is not subject to SRP review criteria, the staff concludes that it does not conflict with current regulatory guidelines. However, an FDA/DC or COL applicant should explicitly state that plant safety will not be compromised in attempting to satisfy refueling, critical maintenance tasks, and other requirements addressed in Appendix D.

CHAPTER 7, "FUELING AND REFUELING SYSTEMS"

1 INTRODUCTION

This chapter of the SER documents the NRC staff's review of Chapter 7, "Fueling and Refueling Systems," of the passive Requirements Document through Revision 5. Revision 5 was submitted to the NRC by letter dated October 1, 1993. Chapter 7 of the passive Requirements Document was prepared, under the project direction of EPRI and the ALWR Utility Steering Committee, by Commonwealth Edison Company; Duke Power Company; General Electric Company; MPR Associates, Inc.; S. Levy Incorporated; Westinghouse Electric Corporation; and EPRI.

1.1 Review Criteria

Section 1 of Volume 1 of this report describes the review criteria used by the staff during its review of Chapter 7 of the passive Requirements Document.

1.2 Scope and Structure of Chapter 7

Chapter 7 of the passive Requirements Document defines the ALWR Utility Steering Committee's requirements for the design of the facilities used to handle and store new and spent nuclear fuel at passive ALWRs.

The key topics addressed in the Chapter 7 review include the spent fuel pool, the fuel pool cooling and cleanup system, the new fuel storage facility, the cask receiving and handling facilities, and the fuel handling system.

1.3 Policy Issues

During its review of Chapter 7 of the passive Requirements Document, the staff did not identify any issues that involve policy questions for the technical areas discussed in this chapter, other than those already identified in the Commission papers listed in Appendix B to Chapter 1 of this report.

1.4 Outstanding Issues

The DSER for Chapter 7 of the passive Requirements Document contained the following outstanding issues:

Open Issues

P.7.0-1 nondestructive testing of the spent fuel pool liner (2.3.1)
P.7.0-2 inservice testing requirements for the spent fuel pool liner (2.3.1)
P.7.0-3 criticality of new fuel in new fuel storage facility (2.3.3)
P.7.0-4 radiological consequences of a fuel handling accident (2.3.9)
P.7.0-5 source term for a fuel handling accident (2.3.9)
P.7.0-6 incorrect reference for water clarity requirement (3.1.1)

Confirmatory Issue

P.7.C-1 quality group classification of components for the new and spent fuel storage racks (2.3.1)

The final disposition of each of these issues is discussed in detail in the appropriate section of this chapter, as indicated by the parenthetical notation following each issue. All outstanding issues identified in the DSER for Chapter 7 have been resolved.

1.5 Vendor- or Utility-Specific Items

The DSER for Chapter 7 contained the following vendor- or utility-specific items. The final disposition of each item is discussed in the appropriate section of this chapter, as indicated by the parenthetical notation following each item.

P.7.V-1 design of the overhead bridge crane (2.3.2)
P.7.V-2 high-radiation areas (2.3.7)
P.7.V-3 reactor disassembly and servicing equipment for boiling water reactors (BWRs) (3.1.2)

The following vendor- or utility-specific items remain and will be addressed during the staff's review of an individual application for FDA/DC. The staff's position on each item is provided in the appropriate section of this chapter, as indicated by the parenthetical notation following each item.

P.7.V-1 design of the overhead bridge crane (2.3.2)

P.7.V-2 radiological consequence assessments of fuel handling and fuel cask drop accidents (2.3.9)

P.7.V-3 reactor disassembly and servicing equipment for BWRs (3.1.2)

2 REQUIREMENTS COMMON TO BWRs AND PWRs

The fueling and refueling systems will include all the facilities for the safe handling and inspection of new fuel, the storage of new and spent fuel, the inspection and repair of spent fuel bundles, and the transfer of new and spent fuel. The structural design requirements are provided in Chapter 1 of the passive Requirements Document. The staff's evaluation of these requirements is given in Chapter 1 of this report.

2.1 General Requirements and Policy Statements

Sections 1.5.2 and 2.2.1 of Chapter 7 of the passive Requirements Document state that the ALWR fueling and refueling systems will be designed so that the reactor can be refueled in 17 days or less. This assumes a 24-month plant operating cycle, which requires the replacement of more fuel assemblies than a 12- or an 18-month cycle. The tasks assumed for the 17-day refueling outage are only those that must be performed each time the reactor is refueled. Additional outage time over plant life is allowed for performing other tasks that are not required each time the reactor is refueled. To meet the 17-day refueling outage requirement, the passive Requirements Document assumes that all shared refueling plant equipment (refueling equipment capable of performing activities not related to refueling) will be available whenever needed, so that there is no delay because of competing use.

One of the objectives of a 17-day (maximum) refueling outage is to reduce the amount of personnel radiation exposure associated with refueling operations. In recent years, refueling operations have accounted for between 6 and 8 percent of the total annual collective radiation dose to plant personnel. EPRI expects that the criteria in Chapter 7 will result in lower radiation doses associated with refueling by permitting fueling and refueling operations to be performed more expeditiously through the use of more automated or remote-controlled processes. Some of the design criteria specified by EPRI to reduce radiation exposure to personnel include the use of permanent refueling seals, the use of integrated head removal features, and the specification of smooth-surface conditions on pool walls to facilitate decontamination.

Section 1.5.6 of Chapter 7 states that man-machine interface (M-MI) requirements specific to fueling and refueling equipment are included in Chapter 7. The staff's review of general requirements for the M-MI, including human factors considerations, is contained in Chapter 10 of this report. One of the goals of Chapter 7, as stated by EPRI, is to eliminate the M-MI problems that have existed in the design of past and current refueling equipment by emphasizing features that simplify the interaction of the operator with the principal operating equipment.

Section 1.5.7 of Chapter 7 describes two different approaches for handling spent fuel and spent fuel casks: the shuttle method and the cask immersion method. EPRI notes that the shuttle method requires further development to determine if it is a viable approach for refueling activities.

In addition, Section 1.5.7 of Chapter 7 specifies two methods to provide assurance that the spent fuel cask will not be dropped into the spent fuel pool or onto safety-related equipment. One of these methods requires the

plant layout design to physically preclude the possibility of a cask drop into the spent fuel pool or onto safety-related equipment. The other method requires that the crane be designed so that it is single failure proof. EPRI states that the first method is preferable and that the single-failure-proof crane should only be used in plants that have building restrictions that prevent the first method. Section 2.3.2.3.2 of Chapter 7 implements this policy by specifying restrictions on crane travel. (Sections 2.3.1.1.4 and 2.3.1.2.4 of Chapter 7 require the pool structure and fuel storage racks to be designed to withstand other dropped loads.)

Section 1.5.10 of Chapter 7 states that the ALWR will be designed to minimize the potential for a refueling pool draindown event.

2.2 Performance Requirements

2.2.1 Nuclear Design Requirements

Section 2.2.2.2.1 of Chapter 7 of the passive Requirements Document specifies that the spent fuel pool storage racks will be designed for subcriticality for all anticipated fuel management programs, including enrichments for 24-month fuel cycles. Specifically, EPRI requires that K_{eff} be no greater than 0.95 in nonborated water at optimum moderator conditions with no normally removable poisons in the fuel. This requirement takes into consideration, for BWR fuel (and some PWR fuel designs) with embedded burnable poison, that the most reactive time in life may not be at beginning of life (fresh fuel). The staff concludes that these requirements are consistent with SRP Section 9.1.2, "Spent Fuel Storage," and are, therefore, acceptable.

2.2.2 System Performance Requirements

Section 2.2.2 of Chapter 7 describes the system performance requirements for the spent fuel pool. In addition to nuclear design requirements, it gives requirements pertaining to capacity and spent fuel pool temperature.

According to Chapter 7, the principal functions of the spent fuel pool, including the spent fuel racks, will be the following:

- provide storage for the following: (1) a full core off-load, (2) spent fuel assemblies until decay heat and radiation generated by the assembly are acceptably low for shipment of the assembly off site or for its transfer to onsite dry storage facilities, and (3) other irradiated reactor core components awaiting shipment for disposal off site

- provide a means for removing the decay heat generated by the spent fuel in the pool and maintain temperatures of the fuel rods well below temperatures at which the cladding could melt or the fuel could be damaged in any way

- safely store spent fuel assemblies in subcritical arrays

- protect personnel by providing shielding against the radiation generated by the stored spent fuel assemblies

- provide space for the reconstitution, repair, and inspections of fuel assemblies and for the consolidation of fuel rods

- provide a sufficient volume of water to ensure adequate cooling of the spent fuel assemblies and shielding of personnel during a station blackout

- provide a fission-product scrub to reduce the radioactivity of iodine released from the pool

Section 2.2.2.3 of Chapter 7 provides detailed specifications for system performance. It requires that the water temperature in the pool be maintained at or below 140 °F with a maximum fuel load. For a normal expected fuel load, EPRI states that this temperature limit should not be exceeded even with a single active failure of the system. However, for a maximum load with a single failure, it specifies a limit of 180 °F, which is still below the boiling point of water.

Section 2.2.3 of Chapter 7 contains specific requirements for the removal of the decay heat generated by the spent fuel and for maintaining high purity of water in the spent fuel pool. Some of the principal functions of the fuel pool cooling and cleanup system (FPCCS) will be to maintain spent fuel pool temperature within acceptable limits, to ensure that there is enough water inventory in the pool so that all the spent fuel assemblies will remain properly immersed, and to maintain water quality in the spent fuel pool.

Regarding the FPCCS, Section 2.2.3.2 of Chapter 7 specifies the following requirements:

- The design flow rate and filtering capability of the FPCCS will be such that the refueling pool water chemistry and clarity will allow an operator to read fuel assembly identification numbers that are 3/8 inch high, 3/16 inch wide, and 1/16 inch thick from the refueling machine when the operators and refueling equipment are ready to move fuel (i.e., designed so that water clarity problems do not cause refueling delays).

- The FPCCS will maintain the water chemistry and clarity of the refueling pool, spent fuel pool, refueling water source (BWR), and in-containment refueling water storage tank (IRWST) (PWR) within the following limits:

  - conductivity less than 3 μsiemens/cm at 25 °C
  - pH between 4.5 and 10 at 25 °C
  - chlorides less than 0.15 part per million (ppm)
  - fluorides less than 0.15 ppm
  - silicas less than 0.10 ppm
  - total suspended solids less than 1.0 ppm

Section 2.2.3.2 of Chapter 7 specifies that the water in the pool should be clear enough so that the spent fuel immersed in the pool will be clearly visible to the operator. To aid in viewing spent fuel, operators may use instruments such as underwater binoculars, floating view plates, and optical fiber viewing devices. To achieve that goal and to prevent materials coming

in contact with water from becoming corroded, the chemistry of the pool water must be rigorously controlled. Also, the potential sources of contamination should be minimized.

In addition, Section 2.2.3.2 of Chapter 7 recommends that EPRI NP-1081, "Refueling Outage Water Clarity Improvement Study," be used as guidance. All these specifications and recommendations comply with SRP Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System," and are, therefore, acceptable.

Section 2.2.3.4 of Chapter 7 of the passive Requirements Document states that the design of the FPCCS will permit operation in the following modes:

- cooling and cleanup of the spent fuel pool
- cleanup of the IRWST (PWR) or the suppression pool (BWR)
- cleanup of the refueling pool
- cleanup of the cask loading pit
- cooling and cleanup of the spent fuel pool simultaneously with one other element of the system
- skimming of the spent and/or refueling pool
- cooling of the spent fuel pool under limiting emergency conditions of single-system failure
- makeup of spent fuel pool inventory

These modes of operation embody the operational requirements specified in SRP Section 9.1.3 and are, therefore, acceptable.

Section 2.2.3.6 of Chapter 7 of the passive Requirements Document states that skimmers, or an equivalent system, will be provided at the periphery of the spent fuel pool and refueling pool to remove foreign material from the water surface, thereby improving water clarity. Skimmers will also reduce the extent of airborne contamination. By improving water clarity, refueling operations can be accomplished more expeditiously, thereby reducing the radiation doses received by refueling personnel. Therefore, the staff concludes that these features are acceptable.

Additional requirements that address the fuel pool cooling and cleanup system are provided in Chapter 8, "Plant Cooling Water Systems," of the passive Requirements Document. The staff addresses these additional requirements in Chapter 8 of this report.

Section 2.2.4 of Chapter 7 of the passive Requirements Document gives the performance requirements for the fuel handling area heating and ventilating system. These requirements address fuel handling area environment, fuel handling area air pressure, area isolation, maintenance access, and air flow arrangement. The staff's evaluation of the fuel handling area heating and ventilation system is given in Chapter 9 of this report.

Section 2.2.5.1 of Chapter 7 of the passive Requirements Document requires that the new fuel storage facility be provided with onsite dry storage for the normal number of new fuel assemblies replaced each refueling outage for fuel cycles up to 24 months. This requirement will ensure that after the initial core load, new fuel will not need to be stored in the spent fuel pool.

Section 2.2.5.2 of Chapter 7 specifies that, as a goal, the new fuel storage facility will be designed so that subcriticality (K_{eff} no greater than 0.95) is maintained for fresh fuel for optimum moderator conditions. In addition, under no conditions will K_{eff} be greater than 0.98. The regulatory position in SRP Section 9.1.1 for new fuel storage is that K_{eff} will be no greater than 0.95 for fully flooded conditions and no greater than 0.98 for low-density (fog or mist) optimum moderator conditions. However, the wording of Section 2.2.5.2, Revision 2, deviates from this position. The requirement to maintain K_{eff} under 0.95 for fresh fuel for optimum moderator conditions is more restrictive and acceptable. However, the requirement that under no conditions shall K_{eff} be greater than 0.98 is somewhat confusing because it is not possible to have K_{eff} greater than 0.95 if it is maintained below 0.95 for optimum moderator conditions. The staff interpretation of the Revision 2 requirements is that EPRI intends for the new fuel storage to be designed with K_{eff} no greater than 0.95 under all conditions including low-density optimum conditions, and, if not achievable, K_{eff} no greater than 0.98 is allowed for the low-density optimum moderator conditions. This would be consistent with the regulatory position of SRP Section 9.1.1 and is, therefore, acceptable.

2.3 System Performance Requirements

2.3.1 Spent Fuel Pool and Related Auxiliary Systems

Sections 2.3.1 and 2.3.3.2 of Chapter 7 of the passive Requirements Document, respectively, state that the requirements in American National Standards Institute/American Nuclear Society (ANSI/ANS) 57.2, "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants," and 57.3, "Design Requirements for New Fuel Storage Facilities at Light Water Reactor Plants," should be considered in the design of all systems and equipment that are associated with the storage of spent and new fuel. For the safety classification of these components, systems, and equipment, ANSI/ANS 57.2 and 57.3 both reference ANSI/ANS 51.1, "Nuclear Safety Criteria for the Design of Stationary PWR Plants," and 52.1, "Nuclear Safety Criteria for the Design of Stationary BWR Plants." ANSI/ANS 51.1 and 52.1 contain the following classification: "New and spent fuel storage racks are classified as seismic Category I but are not required to conform to the quality assurance requirements of 10 CFR Part 50, Appendix B." As discussed in Section 4.3.1 of Chapter 1 in this report, the staff has not completely endorsed these two standards. The staff's position is that the new and spent fuel storage racks for both BWRs and PWRs should be designed, fabricated, constructed, and tested in accordance with the applicable quality assurance requirements of 10 CFR Part 50, Appendix B, in addition to being classified as seismic Category I. In a letter dated December 22, 1989, EPRI stated that 10 CFR Part 50, Appendix B, was applicable to both new and spent fuel storage racks and that Chapter 7 would be modified to clarify this position. The staff identified this as a confirmatory issue in the DSER for Chapter 7. In a letter dated September 23, 1992, EPRI modified its response to this issue by stating that the new industry position was that no revision to Chapter 7 was necessary

because (1) Table B.1-2 in Chapter 1, Appendix B, of the passive Requirements Document commits to Regulatory Guide (RG) 1.29, "Seismic Design Classification," and (2) RG 1.29 states that fuel storage racks should be seismic Category I and, as such, are subject to 10 CFR Part 50, Appendix B. On the basis of this commitment, rather than the requirements in Sections 2.3.1 and 2.3.3.2 of Chapter 7, the staff concludes that acceptable seismic and quality assurance requirements for the ALWR new and spent fuel storage racks have been provided. Therefore, the DSER confirmatory issue is closed.

Sections 2.3.1.1.2 and 2.3.1.3 of Chapter 7 state that the spent fuel pool and the refueling pool will be provided with a suitably sized, leaktight austenitic stainless steel liner plate of welded construction. In addition, Section 2.3.1.3 states that the refueling pool wall liner will be surface finished to reduce the adherence of contamination and increase the efficiency of refueling pool decontamination activities after draining. The liner plate will have ASTM A480/A480M No. 4 surface finish or better, and the liner plate welds will be ground smooth.

In the DSER for Chapter 7, the staff stated that it considered these requirements acceptable because the liner plate will minimize the possibility of leakage of spent fuel pool and refueling pool water, and a smooth surface finish will reduce the crevices where contamination can accumulate. However, in the DSER, the staff concluded that the passive Requirements Document should require that the welded seams of the liner plates be spot radiographed and liquid penetrant and vacuum box examined after fabrication to ensure that the liner does not leak. The staff identified nondestructive testing of the liner as an open issue. In a letter dated September 29, 1992, EPRI stated that the spent fuel pool liner and the refueling pool liner are required to be leaktight; however, the inspection specifications required to verify leaktightness are beyond the scope of the Requirements Document. The staff agrees and concludes that this response is acceptable; therefore, this DSER open issue is closed.

Section 2.3.1.1.6 of Chapter 7 states that a leak chase and collection system will be provided for each underwater seam weld, including vertical seams below the wetted surface, joining the spent fuel pool liner plates and requires that the leak chase system drain to the radioactive waste system and be of a segmented design. The drains will be designed for monitoring leakage rate.

In the DSER for Chapter 7, the staff concluded that these requirements were acceptable because any leakage from the spent fuel pool would be drained to the radioactive waste system. However, operating experience indicates that the pool liner could develop leakage at the welded seams between plates. Therefore, in the DSER, the staff concluded that the passive Requirements Document should specify that the spent fuel pool liner will be inspected (inservice) to ensure that the liner plate welds do not leak and identified inservice testing requirements for the liner as an open issue. In a letter dated September 29, 1992, EPRI stated that the leak chase system required for the spent fuel pool will allow for the identification and repair of liner leaks. The staff concludes that this response is acceptable; therefore, this DSER open issue is closed.

2.3.2 Cask Receiving and Handling Facilities

Section 2.3.2.3 of Chapter 7 specifies that an overhead bridge crane rated for no less than 150 tons should be provided for handling casks and that the design must ensure the safe handling of heavy loads around the reactor vessel and the spent fuel pool and, as a minimum, must meet the provisions of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." The key requirements in this section include the following:

- The crane traveling range and building will be arranged so that the cask, if dropped in any location along its path, cannot damage new or spent fuel, cannot drain the spent fuel pool, and cannot damage any safety-related system or structure, and that the cask itself will not fall in a manner that will result in significant offsite consequences. For the BWR only, if the plant design precludes the use of a separate cask loading pit, a single-failure-proof crane may be used to reduce the probability of a drop along the cask's travel path. However, even for a single-failure-proof crane, interlocks will be used to prevent heavy loads from traveling over the spent fuel pool.

- The overhead crane that will serve the facility where casks will be received will be provided with sufficient lateral movement at the receiving bay level to rotate the cask between horizontal and vertical positions by means of continuous cask rotation with the transporter locked in place. For BWR designs with building size constraints, an acceptable alternative is to move the transporter to provide longitudinal travel and to use the building crane to provide vertical travel. For this alternative, precise alignment and control of the transporter are mandatory. The precision of the movement of the transporter will be consistent with that of the building crane. Movements of the crane and the transporter will be controlled at a station located at the elevation of the cask receiving bay.

- The cask handling crane will have a drive system that will permit continuously variable speed and will include a microdrive feature. A "fast" speed will also be included. Limit switches or other protective devices will be provided to limit the fast speed to those occasions when the crane is unloaded.

The staff concludes that the EPRI requirements for the overhead bridge crane do not conflict with SRP Section 9.1.5, "Overhead Heavy Load Handling Systems." However, there is insufficient detail (e.g., with regard to the design layout, which shows the functional geometric layout of the handling equipment, including the areas of movement over and around the fixed locations of safety-related facilities during the handling of heavy loads, and with regard to the adequacy of operator training, load-handling procedures, and instructions) for the staff to determine if the various handling operations can be performed safely for specific designs. Therefore, the designer or applicant will have to demonstrate in the application for FDA/DC that the design of the overhead bridge crane complies with SRP Section 9.1.5.

2.3.3 Fuel Receipt, Inspection, Storage and Shipment Equipment

Section 2.3.3.2.1 of Chapter 7 states that the new fuel storage facility will be designed so that a fuel assembly cannot be inserted anywhere in the racks other than in the design locations. Provisions are to be made for draining the vault to prevent the accumulation of a fluid moderator, although the nuclear design will assume an optimally moderated fully loaded vault.

In the DSER for Chapter 7, the staff noted that other overmoderated conditions, not just flooding, could occur in the new fuel storage facility as a result of, for example, the actuation of fire suppression systems. Therefore, it concluded that the passive Requirements Document should be modified to stipulate that subcriticality will be maintained under all plant conditions, including foam, mists and sprays, and flooding. The staff considered this an open issue in the DSER. By letter dated September 29, 1992, EPRI stated that the requirements for subcriticality of the new fuel storage facility were contained in Section 2.2.5.2 of Chapter 7. As stated by the staff in Section 2.2.2 of this chapter those requirements are consistent with the regulatory requirements in SRP Section 9.1.1 and are, therefore, acceptable. Therefore, this open issue is closed.

The staff concludes that the design requirements in Chapter 7 of the passive Requirements Document for the new fuel storage facility are consistent with SRP Section 9.1.1 and are, therefore, acceptable. The discussion in Section 2.3.1 above presents the staff's evaluation of the seismic and quality assurance requirements for new fuel storage racks.

2.3.4 Fuel Handling System

Section 2.3.4.1 of Chapter 7 specifies that designs for the fuel handling system and equipment will conform to ANSI/ANS 57.1, "Design Requirements for Light Water Reactor Fuel Handling Systems," except for the seismic classification of the BWR refueling platform assembly. This assembly will be classified as seismic Category II. Other exceptions to ANSI/ANS 57.1 will be specifically justified by the plant designer.

The staff's position relative to the seismic classification of the BWR refueling platform assembly is that either it should be classified as seismic Category I or EPRI should make a commitment to seismically analyze this item to determine if it can withstand the loads resulting from the safe shutdown earthquake (SSE) without any damage to fuel or safety-related equipment. In response to this position, EPRI revised Section 2.3.4.1.1 of Chapter 7 to require that the BWR refueling platform assembly be classified as seismic Category II. As discussed by the staff in Section 4.3.2 of Chapter 1 in this report, Section 4.3.2 of Chapter 1 of the passive Requirements Document now contains a requirement that Positions C.2 and C.4 of RG 1.29; "Seismic Design Classification," be applied to seismic Category II items. A commitment to these positions provides reasonable assurance that the BWR refueling platform assembly will be designed and constructed to withstand the loads resulting from the SSE without any damage to fuel or safety-related equipment and is, therefore, acceptable.

Section 2.3.4.8 of Chapter 7 specifies the following requirements for the fuel handling system materials:

- Structural materials for the fuel handling system components will be selected in accordance with Section 5 of Chapter 1 of the passive Requirements Document, based on a useful life of 60 years.

- All materials for underwater service will be stainless steel or comparable corrosion-resistant alloys in accordance with Section 5 of Chapter 1 of the passive Requirements Document, based on a useful life of 60 years.

- No cadmium-plated fasteners will be used on any equipment placed in any pools.

- The use of paints on fuel handling equipment should be minimized, that is, limited to structures where it would be impractical to use corrosion-resistant materials (e.g., large structural shapes).

The staff concludes that these requirements do not conflict with current NRC regulations or guidance and are acceptable.

2.3.5 Irradiated Fuel Inspection and Repair Equipment

Section 2.3.5 of Chapter 7 specifies that a dedicated space will be provided in the spent fuel pool area for the inspection, repair, and reconstitution of fuel assemblies. Since this space will be large enough to accommodate fuel consolidation equipment, fuel inspection and repair activities will be facilitated. To prevent the spread to other parts of the fuel pool of irradiated particles released during fuel reconstitution, equipment that is state-of-the-art at that time will be used. The method used to segregate the dedicated portion of the pool from the main spent fuel pool area will be dependent on the equipment design and operation and also on the condition of the fuel.

2.3.6 Reactor Disassembly and Servicing Equipment

Section 2.3.6 of Chapter 7 states that the plant designer will provide a stud tensioner design that will minimize the number of lift operations needed to move the reactor vessel stud tensioners and to remove the nuts and washers. Since these operations will be performed in a high-radiation area, this design is intended to result in lower occupational exposures to plant personnel performing these tasks.

2.3.7 Occupational Exposure

SRP Section 12.3, "Radiation Protection Design Features," states that areas that are occupied on a predictable basis during anticipated operational occurrences (such as refueling) should be zoned so that this occupancy results in an annual dose to each of the involved individuals that is as far below the limits of 10 CFR Part 20 as is reasonably achievable.

Section 2.3.1.1.1 of Chapter 7 states that the spent fuel pool should be designed so that the necessary water depth is maintained above the top of the spent fuel assemblies to ensure that the exposure rate to personnel on the spent fuel pool refueling machine will be less than 2-1/2 mrem/hour. This limit is consistent with ANSI/ANS 57.2 and 10 CFR Part 20 and is acceptable.

Section 2.3.1.1.3 of Chapter 7 of the passive Requirements Document states that compression seals will be used for the spent fuel pool gates because such seals are not susceptible to being displaced from the seal cavity (thereby potentially resulting in a partial draining of the pool). In addition, the pool design will not include any drains or permanently connected mechanical or fluid systems whose failure would result in a loss of fuel pool water.

The passive Requirements Document specifies several design features to prevent the partial draining of the fuel pool and the resulting increase in dose rates in the refueling area. The staff notes that in several instances, fuel pools have been partially drained as a result of component failure (such as failure of the transfer canal seal) or improper valve lineup.

To prevent harsh radiological environments above the spent fuel pool, Section 2.3.1.1.1.4 of Chapter 7 states that radioactive non-fuel components in the spent fuel pool will be handled and stored (when not being used) below the normal minimum required water shielding depth.

Section 2.3.1.2 of Chapter 7 states that any fuel pool storage rack locations where the storage of spent fuel could present a radiation hazard to personnel or could otherwise restrict access to locations adjacent to the spent fuel pool will be clearly identified as restricted locations.

The EPRI criteria for the refueling area include several design features for ensuring that occupational exposures are maintained as low as is reasonably achievable (ALARA). Section 2.3.1.1.7 of Chapter 7 states that long-life bulbs will be used in radiation areas so that personnel will not have to enter the areas frequently to replace burned-out bulbs. The spent fuel racks will be designed so that the spent fuel pool floor under the racks can be vacuumed to prevent the buildup of radioactivity in the pool water. Section 2.3.1.3 states that the surface of the refueling pool wall liner will have a smooth finish to reduce the adherence of contamination and to facilitate decontamination of the liner. It also states that a sparger system will be provided around the perimeter of the refueling pool to keep the walls wet to reduce airborne contamination when the pool is being flooded or drained. These features are consistent with the guidelines of RG 8.8 and are acceptable.

Section 2.3.2.4.2 of Chapter 7 specifies that the spent fuel cask servicing and decontamination areas will be close to the loading area to minimize the contamination path created when the wet cask is moved for decontamination. This is consistent with the guidelines of RG 8.8 for minimizing the spread of contamination and is acceptable.

Section 2.3.4.5 of Chapter 7 of the passive Requirements Document describes several design features to help ensure that personnel radiation exposure from operating underwater fuel handling equipment is maintained ALARA. All permanently installed underwater fuel handling equipment will be located so that the levels of radiation from fuel assemblies stored in the spent fuel pool, at the equipment location, are minimized. All underwater fuel handling equipment will be removable and/or replaceable without having to lower the water levels in the refueling pool or in the spent fuel pool. This equipment will be designed to permit easy decontamination. To eliminate concerns associated with radiation streaming through void spaces, all fuel handling

tools used under water will be vented to permit filling with water when submerged in the pool and draining when removed from the pool. These features comply with the guidelines of RG 8.8 and are acceptable.

Section 2.3.6.4 of Chapter 7 of the passive Requirements Document specifies that the reactor building will be arranged to permit the storage of all removable reactor internals under water with resultant radiation levels to personnel maintained at less than 2-1/2 mrem/hr. This feature is intended to maintain radiation doses to personnel ALARA during disassembly and servicing of the reactor. The staff concludes that this feature is acceptable.

Section 4.3.4.4 of Chapter 7 states that a hinged, quick-opening tube closure will be provided for the transfer tube opening inside the reactor containment building in a PWR design. EPRI states that this feature will facilitate the removal and installation of flanges and will result in lower occupational exposure to workers performing these operations. To lower the radiation dose to personnel operating the fuel transfer equipment, the radiation shielding for the fuel transfer system will be designed to limit, to the extent possible, gaps between liner plates, biological shields, and floors. The staff concludes that these design features are intended to reduce radiation exposure to workers during refueling operations and are, therefore, acceptable.

During the transfer of irradiated spent fuel as part of refueling operations, transient or temporary very high radiation levels (e.g., in areas adjacent to the spent fuel transfer tube in PWRs and upper drywell areas in BWRs) could be possible, which could result in potentially lethal radiation doses to personnel. Examples of other plant areas where the potential for transient very high radiation levels may exist are the reactor cavity areas below the reactor pressure vessel and in the seal table room. The sources of very high radiation in both of these areas are the in-core thimbles (for PWRs) and in-core detectors and associated drive cables. These components become activated while inserted in the reactor core and can cause transient high radiation levels in these areas when withdrawn from the core. In the DSER for Chapter 7, the staff stated that EPRI and/or the plant designer should identify all accessible areas where, during normal and anticipated operational occurrences, personnel could receive a radiation dose of 100 rad or more in 1 hour. In addition, the staff recommended that EPRI describe design considerations to ensure that personnel are not exposed to potentially lethal doses of radiation. These design considerations should provide more positive access controls than would be provided by locking area access doors (as is required by 10 CFR 20.203). This was identified as a vendor- or utility-specific issue in the DSER.

EPRI agreed with the staff and, in Revision 3, modified Section 4.2.9.3 of Chapter 6 to require that plant designers identify areas where, during normal operation and anticipated operational occurrences (not related to reactor accident/degraded core conditions), personnel could receive a radiation dose of 100 rad or more in 1 hour. Where practicable, the plant will be designed to preclude accessibility to these areas. For those areas that must be accessible on an occasional basis for maintenance and inspection, the plant designer will ensure that accessibility is limited through the use of positive access controls. The staff concludes that these measures will serve to reduce the possibility of personnel overexposures and are acceptable.

2.3.8 Sabotage Considerations

The staff reviewed the possibility of a saboteur tampering with the crane to damage spent fuel rods and cause a release from the spent fuel pool. The plant layout method of protecting against the drop of a spent fuel shipping cask into the fuel pool, stated in the passive Requirements Document as being preferred, would significantly reduce this possibility. The single-failure-proof-crane method is a less desirable way of protecting against deliberate sabotage.

With regard to tampering resulting in refueling pool draindown, Section 2.3.1 of Chapter 7 specifies compression seals rather than inflatable seals for fuel pool gate seals. Also, Section 2.3.2 of Chapter 4 of the passive Requirements Document requires a permanent seal between the reactor pressure vessel and the refueling canal floor rather than a replaceable gasket. These measures limit the opportunities for draining fuel transfer canals during fuel transfer.

In addition, the ALWR plant arrangement drawings in Chapter 6 of the passive Requirements Document show the spent fuel pool walls as interior walls; however, neither Chapter 6 nor Chapter 7 include that as a requirement. The staff agrees with EPRI's position stated in its August 18, 1989, letter that the spent fuel pool walls are massive enough that a specific requirement that they be located away from the exterior is not essential.

2.3.9 Accident Analysis

In the DSER for Chapter 7, the staff stated that EPRI had not provided an accident analysis for the fuel handling accidents covered in Chapter 7 of the passive Requirements Document. The staff concluded that the passive Requirements Document should address and specify the ALWR design requirements to mitigate (1) the radiological consequences of the release of fission products from irradiated fuel in a spent fuel cask that is postulated to drop during cask handling operations and (2) the radiological consequences of the release of fission products from the fuel in the spent fuel pool or reactor vessel due to the dropping of an object onto irradiated fuel.

In the DSER, the staff also concluded that the passive Requirements Document should establish the source term for a fuel handling accident at an ALWR plant, including, but not limited to (1) the minimum fuel pool water depth for scrubbing airborne radioactive material released from the damaged fuel in the spent fuel pool; (2) the closure time for the containment purge valve and vent valve; (3) the iodine removal efficiency for the fuel handling heating, ventilation, and air conditioning system; (4) the fuel rod gap activity; (5) number of fuel rods involved; (6) potential drop height; and (7) the earliest time after reactor shutdown that fuel handling operations can commence.

The staff identified the source term for a fuel handling accident and the fuel handling accident analysis as open issues in the DSER.

Appendix B to Chapter 1 of the passive Requirements Document contains all the regulatory requirements and guidance applicable to passive ALWRs providing specific requirements for the plant designer. Table B.1-1 of Appendix B commits the ALWR design to the requirements in General Design Criterion

(GDC) 61 in Appendix A to 10 CFR Part 50, except for providing safety-related ac power. Section 4.14 of Appendix B commits the ALWR design to the staff technical provided in RG 1.13, "Spent Fuel Storage Design Basis" (Revision 2). GDC 61 requires that fuel storage and handling systems be designed with appropriate containment, confinement, and filtering systems to ensure adequate safety under normal and postulated accident conditions. RG 1.13 establishes specific design guidance for ventilation and building leakage control capability to limit release of radioactive material under normal and postulated accident conditions. In Section 4.14 of Appendix B to Chapter 1, and Section 1.3 of Chapter 7 of the passive Requirements Document, EPRI requires that the fuel handling area heating, ventiliting, and air conditioning (HVAC) system be designed with appropriate redundancy and charcoal adsorbers and be connected to onsite and offsite ac power without claiming any fission product removal credit for the HVAC system in fuel handling and fuel cask drop accident analyses.

The staff concludes that the EPRI requirements are acceptable because they meet GDC 61 and the guidance in RG 1.13. Therefore, these open issues are closed. However, the staff will perform radiological consequence assessments of fuel handling and fuel cask drop accidents to determine if the design meets the dose reference values in 10 CFR Part 100 during its review of an individual application for FDA/DC or combined license.

Table B.1-2 of Appendix B commits ALWR designers to use the staff technical positions in RG 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," in assessing the ALWR fuel handling and fuel cask drop accidents. EPRI took an exception to the amount of gap activity in the damaged fuel rods at the time of the accident and originally proposed to use 3 percent of fission products in the damaged rods in determining the amount of radioactive material released for assessing the offsite radiological consequences. RG 1.25 specifies 10 percent, and the draft NUREG-1465 "Accident Source Terms for Light-Water Nuclear Power Plants," specifies 5 percent. On the basis of the information in draft NUREG-1465, the staff concludes that 5 percent is the appropriate value for use in NRC analyses, and EPRI has agreed to use the 5-percent value until the final version of NUREG-1465 is published. If the final version of NUREG-1465 supports a lower value, the staff will revise its conclusion.

3 DESIGN-SPECIFIC SYSTEM DESIGN REQUIREMENTS

3.1 BWR System Design Requirements

3.1.1 Fuel Pool Cooling and Cleanup System

Section 3.3.1.1 of Chapter 7 of the passive Requirements Document specifies that, for a BWR, a source of high optical quality water for flooding the refueling pool during refueling will be provided. In the DSER for Chapter 7, the staff noted that this section stated that water clarity requirements were included in Section 2.2.3.2.3. However, those requirements are included in Section 2.2.3.2.2, not Section 2.2.3.2.3. In the DSER, the staff concluded that this apparent typographical error should be corrected and identified this as an open issue. In Revision 4, EPRI revised Section 3.3.1.1 of Chapter 7 to correct this typographical error. The staff concludes that the water chemistry and water clarity requirements in Section 2.2.3.2 of Chapter 7 do not conflict with SRP Section 9.1.3 and are acceptable. Therefore, this open issue is closed.

3.1.2 Reactor Disassembly and Servicing Equipment

Section 3.3.3 of Chapter 7 specifies the following features for BWR designs to facilitate reactor disassembly:

- Drywell head design will incorporate features to facilitate the removal and installation of the head.
- Drywell head and attachment bolts will be stainless steel if water shielding is used.
- Space will be provided on the refueling floor for orderly storage of all items removed or special equipment required during a refueling or service outage.

By facilitating reactor disassembly, EPRI concludes that these features will shorten the refueling outage time and thereby result in lower occupational exposures. The staff concludes that this approach to design is acceptable. However, because there is insufficient detail for it to determine if the various design features are acceptable, the staff will review the design-specific features during its review of an application for FDA/DC.

3.2 PWR System Design Requirements

3.2.1 Fuel Transfer System

Section 4.3.4.1 of Chapter 7 of the passive Requirements Document requires a fuel storage rack capable of storing six fuel assemblies to be included in the refueling pool. In response to the staff comment that the reactivity of these six fuel assemblies should also meet the criterion that K_{eff} be no greater than 0.95, EPRI added the following requirement in Revision 2 to the passive Requirements Document:

The fuel transfer area storage racks shall be designed such that subcriticality is maintained for all anticipated fuel management programs, including enrichments for 24-month fuel cycles. Specifically, K_{eff} shall be no greater than 0.95 in non-borated water at optimum moderator conditions and with no normally removable poisons in the fuel.

The staff concludes that this requirement is acceptable.

3.2.2 Reactor Disassembly and Servicing Equipment

Section 4.3.5.2 of Chapter 7 requires that the reactor vessel be provided with corrosion-resistant stud holes. The threaded stud hole surfaces will be designed so that the repeated removal and installation of the studs will not degrade the corrosion-resistant surfaces.

The staff concludes that these requirements are acceptable because they will prevent borated refueling water from contacting the low-alloy vessel material, which is susceptible to borated water corrosion.

4 CONCLUSION

The staff concludes that the requirements established in Chapter 7 of the passive Requirements Document for the design and arrangement of fueling and refueling systems do not conflict with current regulatory guidelines and are acceptable. However, by themselves, they do not provide sufficient information for the staff to determine if the plant-specific design, operation, and arrangement of the fueling and refueling systems will be adequate. Therefore, applicants referencing the passive Requirements Document will be required to demonstrate compliance with the additional guidance in the SRP, or provide justification for alternative means of implementing the associated regulatory requirements.

Because the requirements in Chapter 7 of the passive Requirements Document pertaining to water chemistry, materials, and materials fabrication supplement the regulatory guidance in the applicable SRP sections, the staff finds them acceptable.

Therefore, the staff concludes that Chapter 7 specifies requirements that, if properly translated into a design in accordance with the NRC regulations in force at the time the design is submitted, should result in a nuclear power plant whose fueling and refueling systems will perform as designed and have all the attributes required by the regulations to ensure that there is no undue risk to the health and safety of the public or to the environment. In addition to complying with existing regulations, such a facility would also be consistent with Commission policies on severe-accident protection and public safety goals.

APPENDIX A
DEFINITIONS AND ACRONYMS

Appendix A of Chapter 7 of the passive Requirements Document contains definitions of terms and acronyms. The staff has provided a consolidated list of acronyms in Volume 1 of this report.

APPENDIX B
PLANT ARRANGEMENT

Appendix B of Chapter 7 of the passive Requirements Document contains a figure for the BWR and one for the PWR, that illustrate designs for the general refueling arrangement that meet the requirements of Chapter 7.

CHAPTER 8, "PLANT COOLING WATER SYSTEMS"

1 INTRODUCTION

This chapter of the SER documents the NRC staff's review of Chapter 8, "Plant Cooling Water Systems," of the passive Requirements Document through Revision 5. Revision 5 was submitted to the NRC by letter dated October 1, 1993. Chapter 8 was prepared, under the project direction of EPRI and the ALWR Utility Steering Committee, by Bechtel Power Corporation; Duke Power Company; General Electric Company; MPR Associates, Inc.; S. Levy Incorporated; Sargent and Lundy Engineers; Stone and Webster Engineering Corporation; Westinghouse Electric Corporation; and EPRI.

1.1 Review Criteria

Section 1 of Volume 1 of this report describes the review criteria used by the staff during its review of Chapter 8 of the passive Requirements Document.

1.2 Scope and Structure of Chapter 8

Chapter 8 of the passive Requirements Document defines the ALWR Utility Steering Committee's requirements for the design of the plant cooling water systems, including component cooling water, service water, circulating water, chilled water, and fuel pool cooling and cleanup systems and the normal power heat sink, at passive ALWRs.

1.3 Policy Issues

During its review of Chapter 8 of the passive Requirements Document, the staff did not identify any issues that involve policy questions for the technical areas discussed in this chapter, other than those already identified in Commission papers listed in Appendix B to Chapter 1 of this report.

1.4 Outstanding Issues

The DSER for Chapter 8 of the passive Requirements Document contained the following open issues:

Open Issues

P.8.0-1 regulatory treatment of non-safety-related active systems (2.2, 3.1, 4.1, 5, and 7)

P.8.0-2 design requirements for the fuel pool cooling system (9)

Confirmatory Issues

None

The final disposition of each of these issues is discussed in detail in the appropriate section of this chapter, as indicated by the parenthetical notation following each issue. All outstanding issues identified in the DSER for Chapter 8 have been resolved.

1.5 Vendor- or Utility-Specific Items

The DSER for Chapter 8 contained the following vendor- or utility-specific items. The final disposition of each item is discussed in the appropriate section of this chapter, as indicated by the parenthetical notation following each item.

P.8.V-1 design requirements for the chilled water system (8)
P.8.V-2 time-delay allowance for fuel pool cooling capability (9)

The following vendor- or utility-specific items remain and will be addressed during the staff's review of an individual application for FDA/DC. The staff's position on each item is provided in the appropriate section of this chapter, as indicated by the parenthetical notation following each item.

P.8.V-1 regulatory treatment of non-safety systems (2.2, 3.1, 4.1, 5, 7, and 8)

P.8.V-2 time-delay allowance for fuel pool cooling capability (9)

2 POLICY STATEMENTS

2.1 Purpose and Scope

Section 1 of Chapter 8 of the passive Requirements Document contains those policies established by the ALWR Utility Steering Committee relating to the plant cooling water systems of ALWR passive plant design. The plant cooling water systems will handle the plant heat loads during normal power operations and reactor shutdown conditions. Chapter 8 specifies the requirements proposed by EPRI for the component cooling water, service water, circulating water, chilled water, and fuel pool cooling and cleanup systems and the normal power heat sink.

2.2 Design Approach

Section 1.5.1 of Chapter 8 of the passive Requirements Document states that the plant cooling water systems for the ALWR passive plant will perform non-safety cooling functions only. This approach is consistent with passive plant policy, which requires that safety functions be performed by inherently passive safety systems. Therefore, active safety-related cooling systems can be excluded and the remaining non-safety systems simplified. EPRI expects that ALWR passive plant cooling water systems (with only non-safety functions) will improve plant reliability because the number of required components (such as pumps, valves, and heat exchangers) and the number of operating alignments will be reduced and the need for technical specifications and limiting conditions for operation (LCOs) on these systems will be eliminated.

In contrast to both the current generation of LWRs and the evolutionary ALWRs, the passive ALWR designs will rely on safety systems that will use the driving forces of buoyancy, gravity, and stored energy sources. These passive systems will supply safety injection water, provide core and containment cooling, and perform other functions. There will be no pumps in these passive safety systems. All valves will be powered by dc electric power from batteries; air-operated; or check valves that are actuated by the pressure differential across the valve. EPRI and the passive reactor vendors propose that these designs not include safety-grade ac electric power.

The passive ALWR designs also will include non-safety-grade active systems to provide defense-in-depth capabilities for reactor coolant makeup and decay heat removal. These systems will serve as the first line of defense to reduce challenges to the passive systems in the event of transients or plant upsets. These active systems will include (1) the chemical and volume control system and the control rod drive system, which provide reactor coolant makeup for the AP-600 and simplified boiling water reactor (SBWR), respectively; (2) the reactor shutdown cooling system and backup feedwater system for PWR decay heat removal and the reactor water cleanup system for BWR decay heat removal; (3) the fuel pool cooling and cleanup system for spent fuel decay heat removal; and (4) the associated systems and structures that will be needed to support these functions, including non-safety-grade onsite standby power sources. In addition, the passive ALWR designs will include non-safety-grade active systems, such as the control room heating, ventilating, and air conditioning system, for mitigating the radiological consequences of an accident. Many of these systems traditionally have been safety-grade systems, but in the

passive plants they will not be designed to meet safety-grade criteria and credit will not be taken for them in the licensing design-basis-accident analyses.

Associated with the new passive design approach, the licensing design-basis analysis will rely solely on the passive safety systems to demonstrate compliance with the acceptance criteria for various design-basis transients and accidents. However, uncertainties remain concerning the performance of the unique passive features and overall performance of core and containment heat removal because no proven operational performance history exists. For example, there are uncertainties in the passive safety systems about the performance of check valves that will operate at low differential pressures provided by natural circulation or gravity injection. These low pressures may not provide sufficient force to fully open sticking check valves (i.e., pumped emergency core cooling systems (ECCSs) are more likely to overcome stuck valves). These uncertainties emphasize the importance of the active non-safety systems being able to provide the defense-in-depth capabilities to prevent and mitigate accidents and core damage. Therefore, the staff must not only review the passive safety systems but also the functional capability and availability of the active non-safety systems to determine if they provide significant defense-in-depth and accident- and core-damage-prevention capabilities.

For those active systems that will perform defense-in-depth functions, the passive Requirements Document specifies performance and systems and equipment design requirements. These include radiation shielding requirements to permit access following an accident, redundancy, the availability of non-safety-grade electric power, and protection against internal hazards, as well as safety analysis and testing to demonstrate system capability to satisfy defense-in-depth considerations. Although it does not provide specific requirements for the reliability of these systems, EPRI, in response to staff questions, has indicated that it is evaluating specific reliability targets and other measures to provide confidence that the passive plants will meet performance requirements and that it will address both passive safety and active non-safety systems.

In addition, technical specification development is a subset of the overall regulatory treatment of the passive designs. The staff is evaluating the need to establish reliability-based technical specifications for passive designs to determine which systems and components (including certain non-safety systems) will require technical specifications and what the parameters of the technical specifications (test frequency, surveillance method, etc.) should be. The reliability assurance program is expected to strongly influence the technical specifications.

Since passive ALWR design philosophy departs from current licensing practices, decisions need to be made concerning the scope of the staff's review of the non-safety systems and of reliance on the passive safety systems. The staff may not require that the active systems meet all the safety-grade criteria, but there should be a high level of confidence to ensure that these systems have adequate capability and availability when called on to perform their defense-in-depth roles. The regulatory treatment of non-safety-related active systems was identified as an open issue in the DSER for Chapter 8. In its letter dated May 26, 1993, EPRI proposed an approach to determine which

systems should receive which kind of regulatory oversight. A key element of this process is to ensure that the passive plant vendors use probabilistic risk assessment methods to examine the reliability of the safety and non-safety systems and ensure that important risk insights are considered in the design. This process and the staff's evaluation of it are described in greater detail in SECY-94-084, which is provided in Annex B of Appendix B to Chapter 1 of this report. In its SRM dated June 30, 1994, also provided in Annex B, the Commission approved the staff's recommendations on RTNSS. In Revision 6, EPRI provided its requirements for implementing the RTNSS process in Section 1.6 of Appendix A to Chapter 1 of the passive Requirements Document. The staff will review implementation of EPRI's process during its review of individual applications for FDA/DC to ensure that the passive plant vendors follow this process satisfactorily. Therefore, this open issue is closed.

2.3 Design Process

Section 1.5.2 of Chapter 8 of the passive Requirements Document specifies requirements to improve the performance of the plant cooling water systems in the areas of thermal and hydraulic analyses of piping and heat transfer systems; optimization of flows, heat transfer areas, and load assignments; degradation of system components as a result of corrosion, silting, and biofouling; and erosion and corrosion considerations.

2.4 Service Water Quality

Section 1.5.3 of Chapter 8 specifies requirements to reduce the corrosive effects of harsh chemistry, silt, and biological fouling on the plant cooling water systems. EPRI specifies that direct service water will not be used for component cooling; raw service water will be treated to reduce the effect of mud, silt, or organisms; materials will be used that offer greater resistance to water chemistry conditions; and provisions will be made to facilitate the inspection of service water piping.

3 KEY REQUIREMENTS

3.1 General Requirements for Plant Cooling Water Systems

In Sections 2 and 3 of Chapter 8 of the passive Requirements Document, EPRI establishes general requirements for the design of plant cooling water systems that are common to the BWR and PWR designs. Additional requirements that are specific to individual cooling water systems are addressed separately in Sections 4 through 9 of this chapter. The key requirements include the following:

- Raw service water will not be used directly for equipment or component cooling. Equipment and component cooling water will be supplied from a closed-loop component cooling water system, which will serve as an intermediate system between equipment or components and the service water system. This will concentrate the fouling and corrosion problem caused by raw service water to one location rather than throughout the plant.

- The plant cooling water systems will not perform any safety cooling function. The portions of the plant cooling water system connected to the standby ac generator are identified in Sections 4 through 9 of this chapter.

- The systems' designs will provide for periodic testing to ensure system adequacy and complete post-maintenance verification of operability.

- The nuclear steam supply component cooling water system, turbine building closed-loop cooling water system, service water system, chilled water system, and circulating water system will be designed with sufficient redundancy of components (i.e., pumps, valves, heat exchangers) to meet ALWR availability goals.

- The normal power heat sink will be a wet-type cooling tower. Both natural and mechanical draft cooling towers will be acceptable. Normal power heat sink capability will be divided among more than one cooling tower or cooling tower segment, with the minimum capacity of each tower segment being determined by the maximum expected decay heat load from the core and spent fuel coincident with a forced plant cooldown to cold-shutdown conditions.

- The nominal flow velocities in the piping of the plant cooling water systems will not exceed 12 feet/second nor be less than 8 feet/second for nominal operating conditions. For other than raw water systems, higher velocities are permitted (about 15 ft/sec).

- Pumps will provide at least a 7-percent margin in head at the pump design point. The head-versus-flow curve will rise continuously from the design point to shutoff.

- The minimum available net positive suction head (NPSH) will be the smaller of (1) 25 percent or (2) 10 feet of water, greater than the required NPSH specified by the pump vendor.

- The raw water systems (e.g., the service water system and the circulating water system) will be evaluated for potential flooding of building compartments containing safety-related equipment or other important equipment as a result of openings or breaks in these systems' piping.

- Adequate filtration and/or silt-removal capability will be provided for raw water systems. Provisions to facilitate periodic cleaning and backflushing will be included.

- Trash racks will be provided upstream of pumps susceptible to damage from large debris. Traveling screens will be installed downstream of the trash racks and upstream of the pumps where the ingestion of fish and vegetation is possible.

- Provisions for physical access and for debris removal from trash racks will not provide a potential path for covert penetration into the protected area.

- Raw water systems will have provisions for biocide injections where discharge is allowed. The plant designer will determine the type of biocide, the concentration of injection, and the frequency of injection on the basis of environmental regulations, makeup water chemistry, and operating conditions, including wet layup and materials. The plant designer will provide a schedule for biocide injections, provisions for neutralizing the discharge, if required, and provisions for filling raw water cooling loops with biocide-treated water before layup. Gaseous chlorine will not be used. If discharge of biocide is not allowed, non-biocide treatments will be used as appropriate. At sites where infestation of microbial growth, such as mussels or hydroids, can occur, provisions will be given in the design for providing thermal backwashing capability.

- Raw water systems will have provisions for the control of scaling. The plant designer will select the method and chemical additions on the basis of environmental regulations, makeup water chemistry, cycles of concentration, and materials and provide a schedule for chemical additions.

- The plant designer will (1) consider the addition of corrosion inhibitors for recirculating (not once-through) raw water systems; (2) determine the need for corrosion inhibitor additions on the basis of environmental regulations, makeup water chemistry, cycles of concentration, and materials; and (3) provide the owner/operator with rationale for corrosion control in the raw water systems.

- Instrumentation will be provided to monitor process conditions and will be sufficient to analyze the performance of the system.

- The plant designer shall provide analyses for the design basis of the plant cooling water systems, modified as necessary for the site, containing the following:

- a parametric cost/performance evaluation that defines system flows versus system heat loads, operating temperatures, time constraints, and relevant pump, piping, valves, and heat exchanger sizing for optimization for various plant operating modes

- the minimum time for detecting component cooling water system-nuclear steam supply system leakage to the service water system and to the normal power heat sink as a function of leakage rate and detection methods

- the NPSH available versus the NPSH required for the pump as a function of flow rate and temperature

- the effects of various operating conditions (such as pumps starting or operating, flow rates, transients) on parameters important for maintaining the system operational

- A detailed hydraulic analysis of all plant cooling water systems will be performed.

- A detailed performance analysis will be performed for all plant cooling water systems to ensure that all design-basis and normal-operating requirements can be achieved for all relevant site envelope conditions.

Section 2.2.6 of Chapter 8 states that the plant cooling water systems will not perform any safety function. Section 2.2.14 of Chapter 8 requires that the nuclear steam supply component cooling water system, turbine building closed-loop cooling water system, service water system, chilled water system, and circulating water system be designed as a single division with sufficient redundancy of components (i.e., pumps, valves, heat exchangers) to meet ALWR goals. As discussed in Section 2.2 of this chapter, EPRI has proposed a process to determine which requirements will be placed on non-safety systems that provide defense-in-depth capabilities for reactor coolant makeup and decay heat removal. This was identified in the DSER as an open issue related to the regulatory treatment of non-safety systems. These systems will serve as the first line of defense in the event of transients or plant upsets to reduce challenges to the passive systems.

The staff will evaluate these systems on a case-by-case basis during its review of an individual application for FDA/DC to ensure that the passive plant designer has followed the process described in EPRI's letter dated May 26, 1993, for determining the appropriate regulatory treatment of non-safety systems. This process and the staff's evaluation of it are described in detail in SECY-94-084, which is provided in Annex B of Appendix B to Chapter 1 of this report. Therefore, this open issue is closed.

The staff concludes that the general requirements in Sections 2 and 3 of Chapter 8 of the passive Requirements Document do not conflict with the guidance in SRP Section 9.2.1, "Station Service Water System," and SRP Section 9.2.2, "Reactor Auxiliary Cooling Water Systems," and are acceptable.

3.2 Equipment Requirements

Section 3.3 of Chapter 8 of the passive Requirements Document contains specific requirements for components of plant cooling water systems, including pumps, heat exchangers, valves, strainers, trash racks and traveling screens, and orifices. General mechanical equipment requirements are included in Section 12 of Chapter 1 of the passive Requirements Document.

3.2.1 Heat Exchangers

Section 3.3.2 contains requirements for plant cooling water system heat exchangers, including the following:

- Heat exchanger fouling factors will be established for each heat exchanger. An appropriate margin in heat exchanger area will be provided to allow for tube plugging.

- Heat exchanger fouling factors will be established by the plant designer by considering conservative predictions of material buildup over the 60-year plant life on the basis of actual system and equipment designs and expected plant operating conditions, including raw service water and component cooling water system chemistry limits. The plant designer will consider a heat exchanger design that includes allowances for fouling, corrosion, and erosion, as well as a planned and programmed tube replacement as an alternative to excessive allowances.

- When not in use, standby heat exchangers wetted by raw water will be provided with a minimum flow or, if isolated, will be drained and filled with a treated water chemistry to prevent corrosion resulting from stagnant water. The minimum flow approach, if used, will be sufficient to prevent accumulation of sediment and debris.

- Heat exchanger tubes will be installed in the tube sheet by a method that minimizes localized crevice corrosion and allows simplified tube removal and replacement.

The staff considers these requirements acceptable because a prediction of material buildup over 60 years will be factored into the heat exchanger sizing.

3.2.2 Materials

General material requirements for passive plant designs are specified in Section 5 of Chapter 1 of the passive Requirements Document. Chapter 2 specifies material specification for the condenser tubes. The following requirements, specified in Section 3.6 of Chapter 8, apply to plant cooling water systems:

- For service water heat exchangers in a fresh water environment, tubes will be Type 304L stainless steel with maximum chloride concentrations less than 200 parts per million (ppm) or the tubes will be Type 316L

stainless steel with maximum chloride concentrations less than 500 ppm. An alternative material with improved corrosion resistance may be specified with the approval of the owner.

- For service water heat exchanger tubes in a salt or brackish water environment, titanium or AL-6XN stainless steel will be specified.

- Test coupons will be used as appropriate, for monitoring performance of system materials.

- The material selection for the condenser tube sheets will be in accordance with Chapter 2 of the passive Requirements Document.

- The service water heat exchanger tube sheets will be specified as follows:

  - For 304L stainless steel tubes, 304L stainless-clad carbon steel tube sheets or solid stainless steel tube sheets will be used.

  - For 316L stainless steel tubes, 316L stainless-clad carbon steel tube sheets or solid stainless steel tube sheets will be used.

  - For AL-6XN stainless steel tubes, solid 304L or 316L stainless steel tube sheets will be used.

  - For titanium tubes, solid titanium tube sheets are preferable. However, solid 304L stainless steel or solid 316L stainless steel tube sheets may be used with titanium tubes.

- Where stainless-clad carbon steel tube sheets are specified, tube-to-tube sheet sealing methods will minimize the potential for replacing the tube sheet cladding after tube replacement.

- Raw water piping installation will permit complete inspection and replacement of sections of the piping for each service water loop and for the circulating water loop during the yearly planned outage schedule (25 days) for an ALWR. The installation of pipe in inspectable trenches will be provided.

- All pipe linings will be capable of being patched and replaced in situ, particularly at weld locations. Pipe linings will be installed only in pipe of 24-inch diameter and larger so that a person can enter the pipe for performing inspections and repairs.

- Copper-based materials will not be specified for raw water systems.

- The materials through which raw water is flowing will be tested with site makeup water. These tests should be performed at typical operating temperatures with water chemistry similar to that expected for the circulating water system and service water system.

- Buried welded steel piping for critical systems will have a nil ductility transition temperature below the lowest expected temperature of the water within the piping.

These requirements are consistent with the general material requirements in Section 5 of Chapter 1 of the passive Requirements Document, which the staff found acceptable.

Section 3.7.3.3 of Chapter 8 of the passive Requirements Document specifies that, as an alternative to corrosion inhibitors, the plant designer may use corrosion-resistant piping and materials and control the closed cooling water chemistry. A filter and deep bed demineralizer to continuously remove corrosion products, dissolved carbon dioxide, and other impurities also will be required. Material selection and water chemistry must be compatible with this alternative.

3.2.3 Raw Water Treatment

Section 3.7.2 specifies the following requirements:

- Raw water systems will have provisions for biocide injections where discharge is allowed. The type of biocide, the concentration of injection, and the frequency of injection will be determined by the plant designer on the basis of environmental regulations, makeup water chemistry, and operating conditions, including wet layup and materials. The plant designer will provide a schedule for biocide injections, provisions for neutralizing the discharge, if required, and provisions for filling raw water cooling loops with biocide-treated water before layup. Gaseous chlorine will not be used. Where discharge of biocide is not allowed, non-biocide treatments will be utilized, as appropriate. At sites where infestation of macrobial growth, such as mussels or hydroids, could occur, considerations should be given in the design for providing thermal backwashing capability.

- Raw water systems will have provisions for the control of scaling. The plant designer will select the method and chemical additions on the basis of environmental regulations, makeup water chemistry, cycles of concentration, and materials. The plant designer will provide a schedule for chemical additions.

- The plant designer will consider the addition of corrosion inhibitors for recirculating (not once-through) raw water systems. The plant designer will determine the need for the addition of corrosion inhibitors on the basic of environmental regulations, makeup water chemistry, cycles of concentration, and materials. The plant designer will provide the owner/operator with rationale for corrosion control in the raw water systems.

The NRC staff considers these requirements acceptable because the required raw water treatment will strongly improve the corrosion resistance of the materials wetted.

4 COMPONENT COOLING WATER SYSTEMS

The component cooling water systems (CCWS) will be closed-loops systems that will provide (1) a method to remove decay heat and waste heat from the reactor, plant systems, structures, and components and transfer the heat to the service water system; (2) protection against leakage of service water into the reactor systems; and (3) protection against release of radiological contamination into the normal power heat sink. The systems will be divided into the component cooling water-nuclear steam supply system and the turbine building component cooling water system.

4.1 Component Cooling Water System-Nuclear Steam Supply System

The component cooling water system-nuclear steam supply system (CCWS-NSSS) will supply cooling water to reactor auxiliary systems and components during normal operation and reactor shutdown conditions. The CCWS-NSSS will consist of a closed-loop division with a minimum of two pumps, two heat exchangers, one surge tank, and associated piping, valves, and instrumentation and controls. In Section 4.2 of Chapter 8 of the passive Requirements Document, EPRI establishes requirements for the design of the CCWS-NSSS. The key requirements include the following:

- During normal operation, one pump and one heat exchanger will be operating to provide the required system flows and heat transfers.
- During normal shutdown, two pumps and two heat exchangers will be in operation.
- The component cooling water electrical loads (e.g., pumps and controls) will be connected to the onsite standby ac power supply.
- Means will be provided to detect and identify leaks from systems that contain radioactivity.
- Instrumentation, with alarms in the main control room, will be provided to monitor radioactivity in the CCWS-NSSS.

Reactor Shutdown Cooling

- The CCWS-NSSS will remove enough residual heat (decay and sensible heat) with all components operable to permit cooling the reactor system in accordance with the performance requirements (e.g., refueling temperature and time) specified in Section 9 of Chapter 3 of the passive Requirements Document.
- With a single active component failure, the CCWS-NSSS will be capable of removing sufficient heat to ensure the reactor system pressure remains less than 135 psig for a BWR and the reactor coolant system temperature less than 350 °F for a PWR.
- Shutdown cooling will be initiated and terminated by operator action.

- The CCWS-NSSS will be designed so that the CCW water will not boil with full-rated residual heat removal (RHR) flow during reactor shutdown cooling.

Safety Pool Cooling

- The safety pool cooling function of the CCWS-NSSS will limit the safety pool (in-containment refueling water storage tank for the PWR and the passive decay heat removal and suppression pools for the BWR) temperatures to limits defined in Section 9 of Chapter 3 of the passive Requirements Document. A single active CCWS component failure will be assumed. Concurrent reactor shutdown cooling need not be assumed.

- This function will be initiated and terminated by operator action.

Fuel Pool Cooling

- The CCWS-NSSS will be capable of removing fuel decay heat from the fuel pool heat exchanger during plant operations and also during plant refueling and maintenance operations.

- This function will be initiated and terminated by operator action.

Miscellaneous Heat Loads

- The CCWS-NSSS, operating in conjunction with the service water system, will provide non-safety-related cooling to reactor non-safety auxiliary systems during all modes of plant operation. The following are typical loads:

  - heating, ventilation, and air conditioning (HVAC) chillers
  - reactor coolant pump motor coolers (for a PWR)
  - sample coolers
  - nonregenerative reactor water cleanup (for a BWR) or chemical and volume control system (for a PWR) heat exchanger
  - reactor coolant drain tank heat exchanger (for a PWR)
  - charging pumps (for a PWR)
  - instrument air compressors
  - waste processing equipment
  - off-gas system (for a BWR)
  - turbine gland coolers (for a BWR)
  - drywell coolers (for a BWR)

In the DSER for Chapter 8, the staff concluded that the design requirements established for the CCWS-NSSS did not meet the guidance in SRP Sections 9.2.1 and 9.2.2 or General Design Criterion 44, "Cooling Water," of Appendix A to 10 CFR Part 50 for ensuring the CCWS-NSSS could perform the traditional safety functions of the system and identified this as an open issue related to the issue of regulatory treatment of non-safety systems.

Section 4.2.2.1 of Chapter 8 states that the CCWS-NSSS will be designed as a closed-loop system that provides cooling water to various non-safety auxiliary components in the reactor and auxiliary buildings.

Section 4.1.2.1 of Chapter 8 states that the CCWS-NSSS will be a non-safety system. As discussed in Section 2.2 of this chapter, EPRI has proposed a process to determine which requirements will be placed on non-safety systems that provide defense-in-depth capabilities for reactor coolant makeup and decay heat removal. These systems will serve as the first line of defense in the event of transients or plant upsets to reduce challenges to the passive systems. This was identified in the DSER as an open issue related to the regulatory treatment of non-safety systems.

The staff will evaluate the CCWS-NSSS during its review of an individual application for FDA/DC to ensure that the passive plant designer has followed the process described in EPRI's letter dated May 26, 1993, for determining the appropriate regulatory treatment of non-safety systems. This process and the staff's review of it are described in detail in SECY-94-084, which is provided in Annex B of Appendix B to Chapter 1 of this report. Therefore, this open issue is closed.

4.2 Turbine Building Component Cooling Water System

The turbine building component cooling water system (TBCCWS) will be a non-safety-related system that will supply cooling water to various auxiliary components in the turbine and service buildings. Section 4.3 of Chapter 8 of the passive Requirements Document specifies that the TBCCWS will be a closed-loop system that will contain a minimum of two pumps, two heat exchangers, and the necessary cross-connecting headers, valves, instruments, and controls to recirculate 100 percent of the required cooling water flow to the individual cooling loads.

In Section 4.3 of Chapter 8, EPRI establishes requirements for the design of the TBCCWS. The staff determined that these requirements, which are mostly based on good engineering practice rather than requirements mandated by regulation, do not conflict with NRC requirements and are acceptable.

5 SERVICE WATER SYSTEM

The service water system (SWS) will be a non-safety-related open system and will be designed to provide cooling water from the normal power heat sink to the tube side of the CCWS heat exchangers that, in turn, will cool various auxiliary components as described by the staff in Section 4 of this chapter.

In Section 5 of Chapter 8 of the passive Requirements Document, EPRI establishes the requirements for the design of the SWS. The key requirements include the following:

- The flow from the cooling tower basin will be supplied by a minimum of two motor-driven pumps. The pumps will be sized so that normal power operation can be supported with one pump out of service (i.e., an installed spare).

- All pumps may be operated to support reactor shutdown cooling. With one pump out of service, the system will be capable of required plant cooling to maintain the reactor in the refueling mode.

- The system will provide a common cooling water supply to both the CCWS-NSSS and TBCCWS heat exchangers or consist of separate subsystems to supply each of the CCWS systems.

- During normal power operation, one or more pumps will operate to remove component heat loads while service water flow remains constant. The standby pumps will be designed to start automatically on low system pressure.

- The use of vertical pumps located at the cooling tower basin with submerged pump suctions, to ensure that adequate NPSH will be available during all operating modes, is permitted if adequate axial support and vibration monitoring equipment are provided.

- Raw service water will not be used directly for equipment or component cooling.

- Adequate filtration and/or silt-removal capability will be provided for raw water systems. Provisions to facilitate periodic cleaning and backflushing, as necessary, will be included.

- Materials for piping, pumps, and heat exchangers will be specified to offer greater resistance to the range of probable water chemistry conditions.

- Material selection for all pump parts wetted by raw cooling water will match the corrosion resistance of the material to the water chemistry. Both operating and stagnant (shutdown) conditions will be addressed. Erosion resistance also will be addressed.

- The plant designer will determine the service water chemistry limits on the basis of the raw water source and the selected raw water treatment in accordance with the common requirements of Section 3.7.2 of Chapter 8.

- Materials in the service water system will be selected in accordance with the results of the above water chemistry determination and as noted in Section 3.6 of Chapter 8.

- Means will be provided to detect leakage into the SWS from the CCWS. A system will be established to take and analyze water samples from the cooling tower basin periodically or when low water level occurs frequently in the CCWS surge tank during normal plant operation. A radiation monitor with a high-level alarm will be placed on the cooling tower blowdown.

- Trash racks will be provided upstream of pumps susceptible to damage from large debris. Traveling screens will be installed downstream of the trash racks and upstream of the pumps where the ingestion of fish and vegetation is possible.

- The raw service water will be maintained at a lower pressure than on the CCWS side during normal operation.

- The heat exchanger shell side will be designed to withstand the maximum pressure that could occur as a result of CCWS pump pressure and static head. As a minimum, the tube side will be designed for static head plus the service water pump shutoff head.

- Instrumentation, with alarms in the main control room, will be provided to monitor process conditions and will be sufficient to analyze performance of the system and to detect problems with the operation of the equipment.

Section 5.3.1 of Chapter 8 states that the SWS may consist of a single closed-loop division. However, the rationale for that section refers the designer to Section 2.2.14 of Chapter 8, which requires the designer to have sufficient redundancy of components to meet EPRI's ALWR availability goals. In the DSER for Chapter 8, the staff stated its concern about the possible use of a single division for the plant cooling water systems that have traditionally been safety-related systems and identified this as an open issue related to the regulatory treatment of non-safety systems.

The staff will evaluate the SWS during its review of an individual application for FDA/DC to ensure that the passive plant designer has followed the process described in EPRI's letter dated May 26, 1993, for determining the appropriate regulatory treatment of non-safety systems. This process and the staff's review of it are described in detail in SECY-94-084, which is provided in Annex B of Appendix B to Chapter 1 of this report. Therefore, this open issue is closed.

6 CIRCULATING WATER SYSTEM

The circulating water system (CWS) will be a non-safety-related open system designed to supply cooling water to the main steam turbine condensers to condense exhaust steam from the low-pressure turbines. The system will consist of piping and large pumps that will provide a low-head, high-flow cooling water supply. Also included will be chemical injection and filters (where necessary) and an optional condenser tube cleaning system.

In Section 6 of Chapter 8 of the passive Requirements Document, EPRI establishes the following key requirements for the design of the CWS.

- The circulating water system will be capable of providing sufficient circulating water to the condenser to support power operations for all expected operating conditions.
- Sufficient circulating water pumps will be installed to permit operation with one pump shut down except during the three hottest months of the year.
- The pumps will be located so that sufficient NPSH is available.
- Trash racks will be provided upstream of pumps to prevent damage from large debris.
- Flooding protection will be provided to ensure that large leaks from the CWS will not result in the loss of all CWS pumps.

The staff concludes that these requirements do not conflict with the guidance in SRP Section 10.4.5, "Circulating Water System," and are, therefore, acceptable.

7 NORMAL POWER HEAT SINK

The primary function of the normal power heat sink (NPHS) will be to provide the source of cooling water to be used as circulating water and service water during power operation. The NPHS will be a non-safety-related open system. The NPHS will consist of cooling tower(s) and basin(s), makeup pumps, makeup intake structure, makeup and blowdown piping, and all associated valves, instruments and controls.

In Section 7.2 of Chapter 8 of the passive Requirements Document, EPRI establishes the requirements for the design of the NPHS. The key requirements include the following:

- The normal power heat sink will be a wet-type cooling tower. Both natural and mechanical draft cooling towers will be acceptable.
- Normal power heat sink capability will be divided among more than one cooling tower or cooling tower segment, with the minimum capacity of each tower segment being determined by the maximum expected decay heat load from the core and spent fuel coincident with a forced plant cooldown to cold-shutdown conditions.
- The cooling tower design will be consistent with cold weather operation at partial load and will consider both icing and non-icing conditions.
- The intake structure for the cooling tower makeup will include a trash rack and traveling screens.
- The cooling tower will be a non-seismic design using the current American Concrete Institute standards (ACI 318 and 334.2) for construction of the concrete superstructure and basin.
- The cooling tower basin will contain provisions, such as access ramps for silt removal vehicles, for the removal of mud and silt.
- The makeup structure will be provided with warm water recirculation for anti-icing protection.
- The cooling tower water treatment and blowdown facilities will be designed on the basis of makeup water chemistry, environmental requirements, evaporation rate, and drift loss.
- Makeup and blowdown lines will be provided with a throttling valve to adjust flow.
- The cooling tower makeup pumps will be capable of taking suction from the makeup source at zero exceedance low water levels.

Section 7.1.1 of Chapter 8 states that the NPHS will be a non-safety system. In the DSER for Chapter 8, the staff stated that it had not yet determined what requirements would be placed on non-safety systems that provide defense-in-depth capabilities for reactor coolant makeup and decay heat removal. Because these systems will serve as the first line of defense in the

event of transients or plant upsets to reduce challenges to the passive systems, the staff identified the requirements for the NPHS as an open issue related to the regulatory treatment of non-safety systems. The staff will evaluate the NPHS during its review of an individual application for FDA/DC to ensure that the passive plant designer has followed the process described in EPRI's letter dated May 26, 1993, for determining the appropriate regulatory treatment of non-safety systems. This process and the staff's review of it are described in detail in SECY-94-084, which is provided in Annex B of Appendix B to Chapter 1 of this report. Therefore, this open issue is closed.

The staff concludes that, while the design requirements established for the NPHS do not meet the guidance in SRP Section 9.2.5 for an ultimate heat sink, they are consistent with the non-safety designation of the system and represent good engineering practice. The staff finds the key requirements in Section 7.2 of Chapter 8 are sufficient to ensure adequate heat removal capability of the NPHS.

8 CHILLED WATER SYSTEMS

In Section 8 of Chapter 8 of the passive Requirements Document, EPRI states that the chilled water system (CDWS) will be a centralized system that will supply various heating, ventilation, and air conditioning (HVAC) and process loads. However, if one centralized system for all buildings is not practical, separate systems will be provided for each building. The system(s) will consist of a refrigeration loop, heat exchangers, chilled water pumps, expansion tanks, and the associated piping and controls.

In Section 8.2 of Chapter 8, EPRI establishes the requirements for the design of the CDWS. The key requirements include the following:

- Performance of the chillers will be based on receiving component cooling water (CCW) at a temperature determined by the plant designer from the requirements of Section 4 of Chapter 8, and on the heat loads and air temperatures specified in Chapter 9 of the passive Requirements Document to support full-power reactor shutdown operation.

- Chilled water will be supplied directly to the HVAC air handling units or, as an alternative, to heat exchangers that will cool CCW supplied to the air handlers.

- Normal makeup will be provided from a demineralized water source.

- The design basis for containment areas will be as follows:

  - CDWS containment penetrations will meet the design requirements of Chapter 5 of the passive Requirements Document.

  - The chilled water pump(s) and refrigeration units required to supply the containment ventilation system on a loss of normal ac power will be automatically supplied with power from onsite standby ac power sources.

  - The CDWS piping and equipment will be supported in accordance with seismic Category II criteria in areas in seismic Category I buildings where required to preclude damage to safety-related equipment in case of a seismic event.

- The chilled water systems will operate in conjunction with the associated HVAC systems to meet the cooling requirements for that specific system.

- The chilled water temperature leaving the chillers will be controlled to a nominal value of 45 °F.

The staff concludes that the design requirements in Section 8 of Chapter 8 of the passive Requirements Document for the chilled water system are commensurate with the system's non-safety classification and similar to the guidance in the Standard Review Plan (SRP, NUREG-0800). However, the chilled water system design requirements are dependent on the outcome of the process described in SECY-94-084 for the RTNSS. Therefore, the degree to which the

guidelines for quality assurance, redundancy, and independence of SRP Sections 3.11, 9.4.1, 9.4.3, 17.1, and 17.2 are incorporated into the design of these systems is dependent on the outcome of the RTNSS process. Therefore, the staff will evaluate this issue during its review of an individual application for FDA/DC.

9 FUEL POOL COOLING AND CLEANUP SYSTEM

Section 9.1 of Chapter 8 of the passive Requirements Document specifies that the fuel pool cooling and cleanup system (FPCCS) will perform the following functions:

- maintain the spent fuel pool temperature within the limits necessary to store and service the spent fuel assemblies removed from the reactor vessel
- maintain the water level in the spent fuel pool within specified limits to ensure that the spent fuel pool assemblies remain cooled and covered at all times
- maintain the quality of the spent fuel pool water and makeup water within the specified limits of chemistry, radioactivity, and clarity
- maintain the water temperature, chemistry, and clarity within acceptable ranges for the in-containment refueling water storage tank (for PWRs) and the suppression pool (for BWRs)

Section 9.2.4.1 of Chapter 8 requires that the system provide cleanup of the spent fuel pool water, refueling cavity water, and safety pool water for removal of particulates (including corrosion products), dissolved and undissolved fission products, and pool water surface dust. Water chemistry will be consistent with the limits specified in Section 5.5 of Chapter 1.

The NRC staff concludes that these requirements are acceptable because the removal of corrosion products from the water and the maintaining of water chemistry within the limits of Section 5.5 of Chapter 1 will ensure that spent fuel pool, refueling cavity, and safety pool will perform as designed.

Section 9.3.2 of Chapter 8 states that the seismic qualification of the spent fuel pool system components will be consistent with the capability of restoring normal cooling following a seismic event. However, as noted in the DSER for Chapter 8, EPRI was not requiring the spent fuel cooling system, component cooling water system, service water system, and normal power heat sink to be designed to seismic Category I requirements. An alternative that is consistent with the loss of normal cooling for 72 hours is seismic Category I piping from the pool to a location in a suitable operating environment where temporary cooling connections can be provided. Included in this alternative would be provisions for assured makeup following a seismic event to fill the pool so that normal cooling could be restored.

The staff found that this approach did not meet the requirements in SRP Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System," which provides two options. The first requires that the cooling portion of the system be designed to seismic Category I, Quality Group C requirements. The second allows a non-seismic Category I, Quality Group C, spent fuel pool cooling system if the fuel pool makeup water system and its source and the fuel pool building and its ventilation and filtration system are designed to seismic Category I requirements and are protected against tornadoes. The staff

concluded that the fuel pool cooling system described in the passive Requirements Document did not conform to either of these approaches; therefore, it was not acceptable. This was identified as an open issue.

In its letter dated September 29, 1992, EPRI stated that the spent fuel pool will be designed with sufficient pool volume to accommodate a loss of normal pool decay heat removal for a period of 72 hours without a need for additional makeup while satisfying the shielding requirements for the fuel. The spent fuel pool cooling system will be analyzed to ensure that for all postulated accidents, the site boundary dose will be below regulatory limits. In Revision 4, EPRI also revised Section 9.3.6 of Chapter 8 to clarify that the designer will have to show by analysis that the offsite radioactive material release will not exceed 10 CFR Part 20 limits for all postulated events. EPRI stated that, after 72 hours, simple means for providing makeup water to the spent fuel pool will permit operating indefinitely in this mode.

The staff reviewed the design of the FPCCS for its capability to cool the spent fuel during all operating conditions. The passive Requirements Document requires seismic qualification for the spent fuel pool system components to ensure their capabilities of restoring normal cooling following a seismic event. Since EPRI has committed to analyze the system to provide evidence that the limits on site boundary dose and the offsite radioactive material release will meet the NRC requirements for all postulated events, the staff concludes that the FPCCS design meets the intent of SRP Section 9.1.3 to ensure that fuel in the spent fuel pool is adequately cooled and, therefore, is acceptable. The staff's evaluation of how EPRI's requirements meet the intent of SRP Section 9.1.3 is provided in Section 4.40 of Appendix B to Chapter 1 of this report. Therefore, this open issue is closed.

In Section 9.3.5 of Chapter 8, EPRI specifies that the electrical power for the FPCCS will be supplied from the normal station auxiliary power during normal plant operation. In the event of loss of all normal ac power, the electrical loads will be connected to the onsite ac power supply. EPRI believes that fuel pool cooling will not be required until some time after normal ac power is lost because of the large heat capacity of the fuel pool water.

The staff agrees in principle with the passive ALWR approach. However, the appropriate time-delay allowance for the fuel pool cooling capability should be evaluated on a design-specific basis because of the flexibility in system design and plant operator actions for an individual ALWR. The staff will review the time-delay allowance for fuel pool cooling capability following a design-basis accident during its review of an individual application for FDA/DC.

10 CONCLUSION

The staff concludes that the EPRI requirements established in Chapter 8 of the passive Requirements Document for the design of plant cooling water systems do not conflict with current regulatory guidelines and are acceptable. However, by themselves, they do not provide sufficient information for the NRC staff to determine if the plant-specific design, operation, and arrangement of the plant cooling water systems will be adequate. Applicants referencing the passive Requirements Document will be required to demonstrate compliance with the additional guidance in the SRP, or provide justification for alternative means of implementing the associated regulatory requirements.

Therefore, the staff concludes that Chapter 8 specifies requirements that, if properly translated into a design in accordance with the NRC regulations in force at the time the design is submitted, should result in a nuclear power plant whose plant cooling water systems will perform as designed and have all the attributes required by the regulations to ensure that there is no undue risk to the health and safety of the public or to the environment.

APPENDIX A
DEFINITIONS AND ACRONYMS

Appendix A of Chapter 8 of the passive Requirements Document contains definitions of terms and acronyms. The staff has provided a consolidated list of acronyms in Volume 1 of this report.

CHAPTER 9, "SITE SUPPORT SYSTEMS"

1 INTRODUCTION

This chapter of the SER documents the NRC staff's review of Chapter 9, "Site Support Systems," of the passive Requirements Document through Revision 4. Revision 4 was submitted to the NRC by letter dated December 18, 1992. Chapter 9 was prepared, under the project direction of EPRI and the ALWR Utility Steering Committee, by Bechtel Power Corporation; Duke Power Company; General Electric Company; MPR Associates, Inc.; S. Levy Incorporated; Sargent and Lundy Engineers; Stone Webster Engineering Corporation; Westinghouse Electric Corporation; and EPRI.

1.1 Review Criteria

Section 1 of Volume 1 of this report describes the review criteria used by the staff during its review of Chapter 9 of the passive Requirements Document.

1.2 Scope and Structure of Chapter 9

Chapter 9 of the passive Requirements Document provides the basic design requirements for the site support systems for ALWRs.

The key topics addressed in the Chapter 9 review include the fire protection system; the environmental monitoring system; the site physical security system; decontamination facilities; compressed air and gas systems; the heating, ventilating, and air conditioning system; and laboratories.

1.3 Policy Issues

During its review of Chapter 9 of the passive Requirements Document, the staff did not identify any issues that involve policy questions for the technical areas discussed in this chapter, other than those already identified in the Commission papers listed in Appendix B to Chapter 1 of this report.

1.4 Outstanding Issues

The DSER for Chapter 9 of the passive Requirements Document contained the following outstanding issues:

Open Issues

P.9.0-1 balance-of-plant fire protection program (3)

P.9.0-2 independence of ventilation systems inside the containment to prevent migration of smoke and hot gases (3.3.1)

P.9.0-3 requirements for smoke-removal capability (3.3.1)

P.9.O-4 safety classification for heating, ventilating, and air conditioning (HVAC) systems (including pump house ventilation system) (8.2)

P.9.O-5 seismic Category II/I criteria for HVAC systems and components (8.2.1 and 8.4.1)

P.9.O-6 use of postfilter instead of high-efficiency particulate air filters (8.2.1)

P.9.O-7 thyroid and beta skin radiation dose requirements (8.2.2)

P.9.O-8 control room habitability following design-basis accident (8.2.2)

P.9.O-9 omission of fans for warehouse areas (8.2.5)

P.9.O-10 omission of cooling coil supplied by chilled water for reactor cavity subsystem (8.4.1)

P.9.O-11 HVAC design for several areas not required to meet requirements specified in the general design criteria, Standard Review Plan (NUREG-0800), or regulations (8.5)

P.9.O-12 use of Uniform Building Code, Zone 2A (8.5)

P.9.O-13 power source and performance requirements following a loss of offsite power (8.5)

Confirmatory Issue

P.9.C.1 guidance on the need for assessing the area interior to security detection equipment (5.2.7)

The final disposition of each of these issues is discussed in detail in the appropriate section of this chapter, as indicated by the parenthetical notation following each issue. All outstanding issues identified in the DSER for Chapter 9 have been resolved.

1.5 Vendor- or Utility-Specific Items

The final disposition of each item is discussed in detail in the appropriate section of this chapter, as indicated by the parenthetical notation following each item.

P.9.V-1 separation of redundant shutdown equipment in the containment (3.3.1)

P.9.V-2 underfloor or ceiling control room cable fires (3.4)

P.9.V-3 security area devitalized during unit shutdown (5.1)

P.9.V-4 security for components that manipulate vital isolation valves (5.2.1)

P.9.V-5 non-safety-related auxiliary systems within the protected area (5.2.1)

P.9.V-6 sabotage vulnerability analysis (5.2.2)

P.9.V-7 charcoal filters in air filtration systems (8.2.4, 8.2.5, 8.3.3, 8.4.1, and 8.4.2)

P.9.V-8 safety classification of fuel facility ventilation supply subsystem (8.2.4)

P.9.V-9 safety classification of PWR auxiliary building ventilation supply subsystem (8.4.2)

The following vendor- or utility-specific items remain and will be addressed during the staff's review of an individual application for FDA/DC. The staff's position on each item is provided in the appropriate section of this chapter, as indicated by the parenthetical notation following each item.

P.9.V-1 design specific details of fire protection program (3)

P.9.V-2 fire hazard analysis (3.2.2)

P.9.V-3 smoke-removal capability (3.3.1)

P.9.V-4 installation of security hardware or fire doors (3.3.1)

P.9.V-5 separation considerations inside the containment (3.3.1)

P.9.V-6 underfloor or ceiling control room cable fires (3.4)

P.9.V-7 security area devitalized during unit shutdown (5.1)

P.9.V-8 security for pressure boundary isolation function (5.2.1)

P.9.V-9 sabotage vulnerability analysis (5.2.2)

P.9.V-10 listing and evaluation of vital equipment (5.2.3)

P.9.V-11 regulatory treatment of non-safety systems (3, 8.2, 8.2.1,8.2.4, 8.4.1, and 8.5)

P.9.V-12 structural integrity of HVAC systems (8.2.2, 8.2.4, 8.4.1, and 8.5)

P.9.V-13 design details of HVAC system (8.2.5)

P.9.V-14 charcoal filters in HVAC systems (8.2.4, 8.2.5, 8.4.1, and 8.4.2)

P.9.V-15 omission of cooling coil supplied by chilled water in reactor cavity system (8.4.1)

P.9.V-16 makeup of PWR auxiliary building ventilation (8.4.2)

2 POLICY STATEMENTS AND KEY REQUIREMENTS

2.1 Policy Statements

Chapter 9 of the passive Requirements Document specifies the requirements for the fire protection system; the environmental monitoring system; the site security system; the decontamination system; the compressed air and gas system; the heating, ventilating, and air conditioning (HVAC) system; and ALWR laboratories. Section 1 of Chapter 9 of the passive Requirements Document contains those policies established by the ALWR Utility Steering Committee related to the design approach to be used in developing the security, fire protection, and compressed air and gas systems for both the BWR and PWR designs.

Section 1.4.1 of Chapter 9 states that EPRI's approach to sabotage protection emphasizes sabotage prevention rather than mitigation. It states that resistance to sabotage will be provided through rugged reinforced-concrete external walls and internal barriers that will restrict access. Protected area barriers, lighting, and intrusion detection systems will be relied on to provide the primary defense against the external threat. Additional resistance will be provided through physically separated, redundant safety systems that will provide high reliability for the prevention of core damage or the mitigation of the consequences of accident sequences. The plant arrangement and the access control features of the plant security system will be compatible with the need to accommodate personnel traffic with minimum delay during normal plant operation and maintenance, and to not impede access or egress under emergency conditions.

In addition, Section 1.4.1 of Chapter 9 originally listed several approaches to providing additional capability for sabotage mitigation that EPRI had "ruled out" for ALWR plants. These included an additional bunkered decay heat removal system, interconnection of systems for additional flexibility, and additional design features solely for sabotage protection as backups to safety systems. The staff concluded that it was premature to rule out consideration by the reactor designer of additional reactor design features to enhance resistance to sabotage for ALWR plants, and EPRI supports this position in the requirements in Section 5 of Chapter 9. In response to staff comments, EPRI added a statement to Section 1.4.1 to require consideration of potential insider and outsider sabotage vulnerabilities and design features to enhance inherent sabotage resistance of the design. This resolves the staff's concern and is acceptable.

Section 1.4.2 of Chapter 9 states that ALWR plants will be equipped with fire protection systems that will, with high reliability, be capable of detecting and suppressing fires that could threaten the health and safety of the public and operating personnel.

Section 1.4.3 of Chapter 9 states that ALWR plants will have the capability to accomplish essential safety functions from control panels external to the control room, should evacuation of the control room be required because of fire or other events, or should a fire damage the control room circuitry that is necessary for the control of safe shutdown systems.

Section 1.4.4 of Chapter 9 states that ALWR plants will include a compressed air and gas system (CAGS) that will minimize air leakage, contamination of air supplies, blockage of air lines, and equipment malfunction. The CAGS will not be designed to meet safety-grade criteria. EPRI states that safe shutdown capability will be ensured by designing equipment served by the CAGS to maintain or assume a safe position on loss of the air supply.

2.2 Site Support Systems - Common Requirements

Section 2 of Chapter 9 of the passive Requirements Document identifies the top-tier requirements applicable to PWR and BWR support systems.

2.2.1 Operability and Maintainability

Section 2.2.1 of Chapter 9 states that components of site support systems will be of a standardized, generic design wherever possible and will be located so as to facilitate testing and maintenance. EPRI indicates that equipment arrangement will provide adequate space for access and replacement. Cableways will allow for ease of access and expansion. Components will be labeled with a unique number and name, consistent with plant drawings.

This section also states that HVAC systems will be capable of maintaining environmental conditions that are compatible with human comfort and with equipment installed in the rooms and compartments served. Access control provided by the security system will permit rapid access to and from vital areas under emergency conditions. The plant designer will ensure that information on system designs is retained and available for later use in the preparation of procedures.

2.2.2 Safety-Related Plant Equipment

Section 2.2.3 of Chapter 9 states that all air-operated safety-related equipment, accumulators, and controls in other ALWR plant systems will be designed for fail-safe operation and will not require an instrument air supply for safe shutdown or accident mitigation. All engineered safety feature systems will be testable to help ensure safe operation in the event of a loss, or a gradual reduction, of the instrument air supply.

2.2.3 Radiation Exposure

Section 2.2.4 of Chapter 9 requires that equipment and facilities subject to radioactive contamination from radiological incidents or normal plant operations be designed to facilitate decontamination and/or to prevent the spread of contamination. Some of the design features that will be used to help ensure that radiation exposure will be as low as is reasonably achievable include the use of curbs around equipment and tanks, floors sloped to drains, coatings that are easily decontaminated, smooth surfaces, and components that are easily accessible. The HVAC system will be designed to minimize the potential exposure of personnel to radioactivity from airborne contamination. To minimize time spent by personnel in radiation areas and to facilitate decontamination, suitable facilities will be located to support maintenance and repair activities performed in high-radiation areas and to support decontamination activities. The materials selected for use in the ALWR will be compatible with the full range of environmental conditions, including

radiation, that may be encountered over the plant life. The specification of nonmetallic parts and electrical components will consider radiation hardening (Chapter 1, Section 12.3.2.1.9) and radiation aging (Chapter 1, Section 6.2.4.5), respectively. The use of radiation-damage-resistant materials in the high-radiation areas should help maximize their service life and reduce the frequency of replacement.

The staff concludes that the above features to minimize personnel radiation exposure and to prevent the spread of contamination comply with the guidelines of Regulatory Guide (RG) 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Reasonably Achievable," and are, therefore, acceptable.

3 FIRE PROTECTION SYSTEMS

Fire protection requirements for nuclear power plants are provided in General Design Criterion (GDC) 3 of Appendix A to 10 CFR Part 50 and in 10 CFR 50.48.

GDC 3 governed fire protection for nuclear power plants and was considered adequate until the fire at the Browns Ferry plant on March 22, 1975. This remains the most serious fire to date in commercial U.S. nuclear power plants. A committee was formed to investigate the fire and to make recommendations on the basis of its findings. One of the recommendations was that specific fire protection guidance should be developed that would supplement the general requirements in GDC 3. The staff published the specific guidance in Branch Technical Position (BTP) APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants," for new plants docketed after July 1, 1976 (revision of Section 9.5.1 of NUREG-75/087 dated May 1, 1976). In August 1976, the staff published Appendix A to BTP APCSB 9.5-1 to provide specific fire protection guidance for those plants that had docketed their applications for construction permits before July 1, 1976. All licensees of operating plants and applicants of plants in various stages of design and construction were asked to review their plants against the guidance in Appendix A to BTP APCSB 9.5-1 and to identify areas of compliance and noncompliance. For identified items of noncompliance, each licensee and applicant was asked to propose modifications to achieve compliance or to show why compliance was not required.

By mid-1979, most plants had complied with most of the provisions of Appendix A to BTP APCSB 9.5-1. However, 18 open issues existed in various combinations at 33 operating plants. The staff then developed 10 CFR 50.48 and Appendix R to 10 CFR Part 50 (published on November 19, 1980, and effective on February 17, 1981) as a means of resolving the remaining 15 open issues (reduced from the original 18 open issues) at plants licensed to operate before January 1, 1979. In addition, the Commission considered sections of Appendix R to be so important that their provisions were required for all plants even if the staff had previously approved the design in those areas. The three sections of Appendix R that applied to all plants were III.G ("Fire Protection of Safe Shutdown Capability"), III.J ("Emergency Lighting"), and III.O ("Oil Collection System for Reactor Coolant Pump"). After 10 CFR 50.48 and Appendix R to 10 CFR Part 50 were published, the staff revised BTP APCSB 9.5-1, which became BTP CMEB 9.5-1 (July 1981, as part of NUREG-0800 (Standard Review Plan, SRP)) to include the provisions of Appendix R so as to give additional guidance to those plants that had docketed their applications for construction permits before July 1, 1976, and that were still being completed and were preparing for operating licenses.

It is important to note that this subsequent fire protection guidance for operating plants, as well as for plants still being constructed, is derived and represents deviations from the original guidance (BTP APCSB 9.5-1, May 1, 1976) developed for new plants. The intention has always been that when any advanced reactor designs were proposed, fire protection would be provided on the basis of the best technology available, not on the basis of methods allowed for plants already operating or in advanced stages of design and construction.

On this basis, the staff evaluated the criteria in the passive Requirements Document for the fire protection system against the criteria of SRP Section 9.5.1 (BTP CMEB 9.5-1, July 1981) and supplemental guidance issued by the Commission. Three examples of such supplemental guidance are (1) Generic Letter (GL) 81-12, which contains information on safe shutdown methodology; (2) GL 86-10, which contains important technical information, such as conformance with National Fire Protection Association codes and standards; and (3) the Commission's staff requirements memorandum (SRM) dated June 26, 1990.

In SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," the staff concluded that fire issues that have been raised through operating experience and through the External Events Program must be resolved for evolutionary ALWRs. To minimize fire as a significant contributor to the likelihood of severe accidents for all new reactor designs including the EPRI ALWR passive plant, the staff proposed that the Commission enhance NRC's current guidance. In its SRM of June 26, 1990, on SECY-90-016, the Commission approved the staff's position regarding review criteria for fire protection design, as discussed in SECY-90-016 and supplemented by the staff's April 27, 1990, response to comments by the Advisory Committee on Reactor Safeguards. In SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," the staff determined that there were no unique features of the passive plants that would preclude evaluating those designs against the enhanced fire protection criteria. The designers of passive plants must demonstrate that safe shutdown of their plants can be achieved, assuming that all equipment in any one fire area has been rendered inoperable by fire and that reentry to the fire area for repairs and for operator actions is not possible. The control room is excluded from this approach, subject to the need for an alternative shutdown capability that is physically and electrically independent of the control room. Fire protection for redundant shutdown systems in the reactor containment building should ensure, to as great an extent as possible, that one shutdown division will be free of fire damage. Consideration should be given to safety-grade provisions for the fire protection systems to ensure that the remaining shutdown capabilities are protected. In addition, the designers should demonstrate that smoke, hot gases, or the fire suppressants will not migrate into other fire areas to the extent that safe shutdown capabilities, including operator actions, could be adversely affected. It is anticipated that this will be accomplished, in part, by providing separate ventilation systems for redundant trains. The staff evaluated the criteria in the passive Requirements Document for the fire protection system against these criteria. The staff will review specific design details during its review of an individual application for FDA/DC.

General Evaluation

EPRI has generally followed the NRC's concept of defense-in-depth with regard to fire protection. The three steps of defense-in-depth and EPRI's implementation of these steps follow:

(1) Reduce the possibility of fire starting in the plant - EPRI has used fire-resistant and fire-retardant materials in its design to minimize and isolate fire hazards.

EPRI will use either low-voltage or fiber optic multiplexed circuits in its design, thus eliminating the need for cable spreading rooms and substantially reducing the amount of combustible cable insulation and higher voltage ignition sources in the control room.

(2) Detect and suppress fire promptly - EPRI has provided automatic detection and a suitable mix of automatic and manual fire suppression capability in its design.

(3) Ensure that any fire that might occur will not prevent safe shutdown of the plant even if fire detection and suppression efforts should fail - EPRI has attempted to ensure this. A detailed staff evaluation of the effectiveness of this approach is provided in the following sections.

The fire protection program described by EPRI is intended to protect safe shutdown capability, prevent the release of radioactive materials, minimize property damage, and protect personnel from injury as a result of fire.

EPRI considered not only the three aspects of defense-in-depth outlined above, but also such features of general plant arrangement as access and egress routes, equipment locations, structural design features that separate or isolate redundant safety-related systems, floor drains, ventilation, and construction materials.

EPRI has used applicable National Fire Protection Association codes and standards in its design and layout of the facility. An ALWR designer or applicant will be required to identify any deviations from these codes and standards and to describe in the fire hazard analysis the deviations and measures taken to ensure that equivalent protection is provided for a plant-specific design.

In the DSER for Chapter 9, the staff stated that it was concerned about the possible lack of adequate fire protection for systems located in the balance of plant (BOP) because the passive Requirements Document did not describe the fire protection program for the BOP. Therefore, the staff could not determine the potential effects of fire. This was identified as an open issue in the DSER.

In its letter dated September 29, 1992, EPRI responded that the ALWR program position is to require fire hazard analyses of non-safety areas of the plant as described in Section 3.2.2.2 and fire protection systems as described in Section 3.3.3 from an investment protection standpoint. EPRI stated that the NRC staff review of fire protection systems for non-safety areas of the plant should not be required. It believes that NRC staff interest in the fire protection details for BOP areas is related to the treatment of non-safety systems. Therefore, it proposed that this issue be resolved as part of the overall question of regulatory treatment of non-safety systems.

The staff expects, as a minimum, that fire protection will be provided as specified in SECY-90-016 and SECY-93-087. SECY-90-016 states, "The ALWR Requirements Documents indicates that fire protection will be as specified in 10 CFR 50.48 and Appendix R." Section 50.48 of 10 CFR states that BTP APCSB 9.5-1 addresses fire protection for safety and non-safety areas. BTP APCSB 9.5-1 has been superseded by SRP Section 9.5-1, "Fire Protection

Program." SRP Section 9.5-1 also specifies the fire protection that is expected in non-safety areas of the plant (e.g., spent pool area, radwaste and decontamination areas). It is expected that fire protection features will be provided as specified in SRP Section 9.5-1. In addition, the plant should be designed so that the potential effects of a fire (including the migration of smoke) for systems located in the BOP should not prevent achieving and maintaining a safe shutdown condition.

On the basis of its review of EPRI's position, the staff has determined that the regulatory treatment of the fire protection program for the BOP will be in accordance with the resolution of Policy Issue III.A, "Regulatory Treatment of Non-Safety Systems (RTNSS) for Passive ALWRs," which was identified in Annex A of Appendix B to Chapter 1 of the DSER. In SECY-93-084, provided as Annex B of this appendix, the staff describes and evaluates EPRI's process for determining RTNSS. In the process, the ALWR designer will identify all risk-significant non-safety systems, structures, and components (SSC) functions and their corresponding reliability/availability (R/A) missions. Furthermore, the designer will propose regulatory treatment appropriate to provide confidence that the identified R/A mission will be achieved. In its SRM dated June 30, 1994, the Commission approved the staff's recommendations on RTNSS. The staff will evaluate the results of this process during its review of an individual application for FDA/DC to determine if enhanced regulatory oversight is required for these systems. Therefore, this open issue is closed.

3.1 System Definition

EPRI states that the scope of Section 3 of Chapter 9 of the passive Requirements Docuemnt encompasses the systems required for protecting ALWRs from fire, including assurance that the plant can achieve and maintain safe shutdown in the event of a fire. In addition, EPRI states that the fire protection systems will ensure personnel safety, protection of property, and continuity of power production. Systems included in Section 3 of Chapter 9 are fire and smoke detection systems and automatic and manual fire suppression systems. Building structural and physical arrangement features provided to enhance fire protection are evaluated by the staff in Chapter 6 of this report. The staff's evaluation of the effect of fire protection system actuations on electric power systems is given in Chapter 11.

3.2 Performance Requirements

3.2.1 General

Section 3.2.1 of Chapter 9 of the passive Requirements Document states that the plant fire protection system will be designed, installed, and tested in accordance with Section 9.5.1, "Fire Protection Program," of the SRP (NUREG-0800). The SRP includes BTP CMEB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants," dated July 1981.

3.2.2 Fire Hazard Analysis

Section 3.2.2 of Chapter 9 states that the plant designer will perform a fire hazard analysis, as described in SRP Section 9.5.1, that will be expanded to include additional plant areas containing non-safety-related equipment.

The staff considers this commitment to be acceptable for the passive Requirements Document. However, it will evaluate the fire hazard analysis proposed for individual plants during its review of an individual application for FDA/DC. As stated previously, the staff will require that fire protection for the ALWR passive plant be provided on the basis of the best technology available. Therefore, using only the guidance of SRP Section 9.5.1 when developing the fire hazard analysis may not be sufficient for a passive plant design.

3.3 System Features

3.3.1 Protection of Redundant Safety Divisions

Section 3.3.1 of Chapter 9 of the passive Requirements Document states that safe shutdown equipment will be protected from fire damage by complete separation of redundant divisions of safe shutdown equipment. Separation will be by use of 3-hour-fire barriers.

In the DSER for Chapter 9, the staff stated that it was concerned that smoke, hot gases, and fire suppressants could migrate to other fire areas to the extent that they could affect safe shutdown capability, including operator actions.

The staff stated that the commitment by EPRI to use 3-hour-fire barriers for separating safe shutdown systems outside the containment complied with the review criteria in SRP Section 9.5.1 and was acceptable. However, EPRI needed to demonstrate sufficient independence of ventilation systems to prevent smoke and hot gases from migrating between redundant trains and adversely affecting operator actions. This was identified as an open issue in the DSER.

In its letter dated September 29, 1992, EPRI stated that it is the intent of the ALWR program to fully commit to designs that will prevent the migration of smoke and hot gas from one fire area containing safety-related equipment to other fire areas containing redundant safety-related equipment. Specific provisions to achieve this objective are described in the Chapter 9 of passive Requirements Document as follows:

- Section 3.3.1.1 requires separation of redundant trains of safety-related equipment by 3-hour-fire barriers (except for the main control room and inside the containment).

- Section 8.2.1.1.23 requires HVAC systems to be compatible with fire zone boundaries.

- Section 8.2.1.1.28 requires controls for redundant HVAC trains to be integrated with the plant fire detection and protection plan to prevent smoke and hot gases from migrating between redundant HVAC trains.

- Section 8.2.1.1.4 requires duct penetrations through fire barriers to have dampers (at the fire barrier) with a rating equivalent to the fire barrier rating.

- Section 8.2.2.3.1 requires Class 1E switchgear rooms to have redundant cooling systems separated from each other by physical barriers.

- Section 8.2.2.4.1 requires Class 1E battery rooms to have separate exhaust systems.

The staff finds that the passive Requirements Document requires designs that will prevent the migration of smoke and hot gas from one fire area containing safety-related equipment to other fire areas containing redundant safety-related equipment. Therefore, this open issue is closed.

Requirements for smoke-removal capability were not considered in Section 3.4 of Chapter 9. This was identified as an open issue in the DSER.

In its letter dated September 29, 1992, EPRI responded that the requirements pertaining to the capability to remove smoke from the main control room were defined in Section 8.2.2.1.3 of Chapter 9.

In Revision 2, EPRI added a requirement in Section 8.2.1.1.28 to integrate the controls for redundant HVAC trains with the plant fire detection and protection plan to prevent smoke and hot gases from migrating between redundant HVAC trains. The staff will evaluate the capability to remove smoke resulting from a fire in the nuclear power plant during its review of an individual application for FDA/DC in accordance with SRP Section 9.5.1. Therefore, this open issue is closed.

In a letter dated June 8, 1989, the staff requested that EPRI provide a description of how security hardware will be installed on fire doors so as to not compromise the fire rating of the doors. In its October 19, 1989, letter, EPRI responded that this description entails detailed engineering instructions that are beyond the scope of the passive Requirements Document. Therefore, the staff will review details of the installation of security hardware on fire doors during its review of an individual application for FDA/DC to ensure that the fire rating of the doors has not been compromised.

EPRI states that safe shutdown systems inside the containment will be protected from fire by spatial separation and by limiting the combustible loading, combined with detection and suppression as determined by the fire hazard analysis. The staff finds this commitment acceptable. However, caution must be exercised in the actual system layout to ensure that line-of-sight exposure between components requiring separation does not exist and that a sufficient labyrinth exists between the separated components to ensure that fire does not spread. Since the containment is considered to be a single fire area, the separation of redundant shutdown equipment, including associated cables, should be such that one shutdown division will remain free of fire damage. In the DSER for Chapter 9, the staff stated that it would evaluate the specific design during its review of an individual application for FDA/DC to ensure these considerations were included.

In its letter dated September 29, 1992, EPRI responded to the DSER stating that its position was that fire hazards inside the containment were very low for ALWRs because of design improvements. The letter stated that plant arrangements need to have flexibility to locate containment wall penetrations for redundant trains in adjacent 90 degree quadrants with only distance providing the separation. There will be cases where "line-of-sight" can be expected to occur between redundant trains and should be permitted. Barriers or vestibules should not be imposed to separate the redundant systems unless

they are very close. EPRI felt that the NRC proposal to add labyrinths or physical barriers just for separation would defeat some of the important ALWR objectives related to improved access for maintenance and operability of components and resulting reduction of radiation exposure to personnel. However, EPRI agreed that this issue could not be fully resolved until the staff reviewed the vendors' plant arrangement inside the containment in conjunction with the containment fire hazard analyses.

The staff does not expect the passive ALWR design to compromise fire protection. It expects fire protection to be integrated into the ALWR design inside the containment so that it will not affect other important design objectives such as improved maintenance and the reduction of radiation exposure to personnel. Therefore, the staff's position remains unchanged. It will evaluate the specific design during its review of an individual application for FDA/DC to ensure that adequate separation considerations are included.

3.3.2 Component Replacement

Section 3.3.2.1 of Chapter 9 states that the fire protection equipment will be designed and installed to facilitate replacement necessitated by aging, early failure, or obsolescence.

3.3.3 Extended Fire Protection Coverage

Section 3.3.3 of Chapter 9 states that, for purposes of investment protection, the turbine-generator and associated areas described in National Fire Protection Association (NFPA) 803, "Fire Protection for Light Water Nuclear Power Plants," will be protected in accordance with NFPA 803. All areas of the plant provided with automatic fire suppression systems will also be provided with manual backup fire suppression capability. Areas not required to have automatic fire suppression systems will be provided with manual fire suppression capability on the basis of the fire hazard analysis. In areas where equipment contains a sufficient quantity of combustible materials, a fixed fire suppression system, designed in accordance with NFPA 803, will be provided. In areas where the fire hazard does not warrant a fixed suppression system, automatic fire detection will be provided. Automatic fire suppression systems will be provided for warehouses containing high-value equipment and combustible materials that are critical for power generation or that constitute a fire hazard for other buildings important to power production.

Section 3.3.3.5 of Chapter 9 states that hydrants will be provided in sufficient number to provide (1) two streams for every part of the interior of any building not covered by standpipe protection and (2) hose stream protection to every part of each building. Each hydrant will be provided with its own isolation valve.

Ventilation filters that collect combustible material and are potential fire hazards will be considered in the fire hazard analysis. Fire suppression will be provided as necessary, in accordance with Section 3.3.3.6 of Chapter 9.

The staff concludes that the requirements of this section are consistent with the enhanced fire protection criteria in SRP Section 9.5.1 and supplemental guidance issued by the Commission as discussed above and are, therefore, acceptable.

3.4 Component Features

Section 3.4.1 of Chapter 9 specifies that pre-action sprinkler systems will be used wherever the consequences of leakage or inadvertent sprinkler system operation are great.

Section 3.4.2 of Chapter 9 states that two or more fire pumps will be provided so that 100 percent of the design capacity of fire pumps will be available if the largest pump is inoperable or offsite power is lost. At least 100 percent of design fire pump capacity will be supplied by a diesel-driven fire pump(s). Automatic fire pump start controls will be provided to maintain pressure in the fire main.

The staff concludes that the requirements of this section are consistent with the enhanced fire protection criteria in SRP Section 9.5.1 and supplemental guidance issued by the Commission as discussed above and are, therefore, acceptable.

Section 3.4.3 of Chapter 9 states that fire protection water will be from a treated, filtered, dedicated source to reduce contamination by biological fouling and debris. One of the sources of fire water supply, including associated storage, the pump suction piping, the pump, the pump discharge piping to the fire main, and the portion of the fire main necessary for manual backup protection of safe shutdown capability following an earthquake will be designed to seismic Category I criteria.

Section 3.4.4 of Chapter 9 states that the fire main pressure will be maintained at least 10 psig above the set pressure for automatic starting of the fire pumps. A hydropneumatic system will not be used for this purpose. Although other methods may be used, EPRI recommends the use of a jockey pump.

Section 3.4.5 of Chapter 9 states that the use of carbon dioxide fixed flooding systems will be minimized because of potential hazards to personnel, thermal shock effects, and the need for the compartment boundary to be leaktight. These systems, where used, will be provided with air test connections for surveillance of nozzles and headers. Fire protection suppression system relays for carbon dioxide fixed flooding systems will not be seismically sensitive.

Section 3.4.6 of Chapter 9 states that controls for sprinkler systems protecting cable trays will be sufficiently sensitive to actuate in the event of a cable tray fire.

Section 3.4.7 of Chapter 9 states that portable fire extinguishes will not be located in highly radioactive areas unless the fire hazard analysis indicates that a specific need exists. These fire extinguishers will be located in unobstructed areas and will be readily accessible to highly radioactive areas. Hose stations will be located outside highly radioactive areas except as necessary to provide a minimum of one hose stream, with 100 feet of hose, to any location that will contain or could present a hazard to safety-related equipment.

Section 3.4.8 of Chapter 9 states that the plant designer will consider the specific combustion products, type of hazard, fire load, and the effects of

humidity, air velocity, temperature, air pollution, radiation, and pressure when selecting the type of fire detector to be used for each application and the location of detectors. Periodic maintenance requirements and personnel radiation exposure also will be considered when selecting the location of detectors.

Section 3.4.9 of Chapter 9 states that cable installations in the control room underfloor or ceiling spaces will be designed to eliminate the need for automatic fire suppression in these areas or in raceways. Also, access will be provided for the use of manual fire extinguishers to reach any portion of the underfloor or ceiling spaces containing any significant fire load. Although the staff considers this commitment to be acceptable in intent, it will evaluate in detail the issue of cables in the underfloor and ceiling spaces in the control room during its review of an individual application for FDA/DC.

Section 3.4.10 of Chapter 9 states that the plant designer will consider potential corrosive effects of fire suppression agents on plant equipment in the selection of agents and in the design of exposed equipment.

Section 3.4.12 of Chapter 9 states that diesel generator areas will be protected with automatic foam sprinklers and that foam hose reels and carbon dioxide hand-held fire extinguishers also will be provided in these areas.

The staff concludes that the requirements for the component features discussed in this section are consistent with the enhanced fire protection criteria in SRP Section 9.5.1 and supplemental guidance issued by the Commission as previously discussed and are, therefore, acceptable.

4 ENVIRONMENTAL MONITORING SYSTEM

4.1 Scope and Functions

Section 4.1 of Chapter 9 of the passive Requirements Document states that the environmental monitoring system will include the systems and equipment that will provide the data necessary for controlling plant releases and for assessing the plant effluent releases to the environment. Included will be the systems and equipment for monitoring meteorological data, water quality, solid waste, and offsite radiation levels. The information from the monitoring systems will be used (1) before plant construction to establish a baseline for licensing purposes, (2) to predict the effects of radiological effluents on the surrounding environment, and (3) to collect data needed to assess the environmental effect of cooling water discharges and cooling tower blowdown and drift.

The staff's evaluation of the plant meteorological system and the water quality monitoring requirements follows. The requirements for radiation monitors, including the monitoring system for the solid radioactive waste processing system, are evaluated in Chapter 12 of this report.

4.2 Performance Requirements

Section 4.2.1 of Chapter 9 states that onsite instrumentation will be provided for monitoring wind speed and wind direction and the vertical temperature gradient. Redundant equipment will be provided as necessary to achieve a 90-percent joint annual recovery of wind-speed, wind-direction, and atmospheric stability parameters in accordance with RG 1.23, "Onsite Meteorological Programs (Safety Guide 23)." Sampling and recording intervals will be in accordance with this regulatory guide. Offsite equipment will be provided as necessary. Provisions will be made for continuously collecting meteorological data for 24 months. Real-time, continuous display of monitored variables will be provided in the control room and emergency response facilities.

With regard to onsite meteorological instrumentation, the passive Requirements Document specifies the use of RGs 1.23 and 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident."

The staff concludes that the use of RGs 1.23 and 1.97 for defining the system's capability to monitor wind speed and wind direction and to determine atmospheric stability is acceptable.

Section 4.2.2 of Chapter 9 states that necessary sensors and equipment will be provided to monitor and access those parameters required by State and Federal environmental agency regulations.

4.3 System Features

Section 4.3.1 of Chapter 9 states that the primary meteorological tower will have instrumentation sensors at the 10-meter level and at a height approximately the level of routine radioactive nuclide release from the plant vent. The equipment at the 10-meter elevation will monitor the standard deviation of

vertical wind speed as a means of measuring atmospheric stability. Visibility, dew point, and solar radiation measurements will be provided if necessary to support cooling tower operation. The section further states that a third set of sensors will be provided above the routine release height if the height of the vent is significantly greater than 60 meters.

Section 4.3.2 of Chapter 9 states that an additional tower(s) will be provided if necessary because of site topography, local meteorology, and building conditions.

Section 4.3.3 of Chapter 9 states that towers will be sited to provide representative measurements of the transport and diffusion characteristics of the site. Generally, surrounding obstacles will be located at a distance at least 10 times their height.

Section 4.3.4 of Chapter 9 states that controls for meteorological monitoring instrumentation will be located in a weatherproof, lightning-protected building. The building will be provided with heating, ventilating, and air conditioning as necessary to meet the environmental criteria of the enclosed equipment.

Section 4.3.5 of Chapter 9 states that sensors, instrumentation, and cabling will be provided with lightning, power surge, and ice protection. Instrument cables will be shielded. At sites on coastal locations, sensors will be provided with a means of preventing or removing salt buildup.

Section 4.3.6 of Chapter 9 states that the environmental monitoring system (EMS) will be energized from a uninterruptible power source from the onsite ac system and will meet RG 1.97. Offsite EMS equipment will be provided with backup battery power.

4.4 Instrumentation and Controls

Chapter 10 of the passive Requirements Document specifies the general requirements applicable to EMS instrumentation and controls, which the staff evaluates in that chapter of this report. Standard, commercially available equipment will be used.

Section 4.4.2 of Chapter 9 of the passive Requirements Document states that equipment of generic design will be used where possible to reduce variation in operation, calibration, and spare parts and reduce dependence on vendors.

Section 4.4.3 states that the plant designer is required to supply documentation for EMS equipment. The information will include the following: function of each instrument, required indication and alarm, required control actions, range, anticipated background radiation, accuracy, response time, and failure mode. Instrument vendors will be required to supply information regarding extremes in environmental conditions as well as operation and maintenance manuals.

Section 4.4.4 of Chapter 9 states that testing and calibration requirements will be derived from manufacturer's recommendations and from guidelines specified in RGs 1.23 and 1.97.

Section 4.4.5 of Chapter 9 states that EMS instrument ranges will be consistent with alarm setpoints and with RGs 1.23 and 1.97. Separate instruments with overlapping ranges will be used if the required range would result in inadequate sensitivity under normal conditions.

4.5 Data Management

Section 4.5 of Chapter 9 states that a computer-based data management and surveillance system will be provided for input and recall of all EMS and laboratory data. The system will be capable of providing daily evaluations of radiological conditions and of tracking and plotting trends.

All EMS instrument channels and the data management system will be designed to interface with the plant data transmission and data base systems for storing and sharing information.

4.6 Conclusion

The staff concludes that the requirements of Section 4 of Chapter 9 of the passive Requirements Document are consistent with the regulatory criteria in RGs 1.23 and 1.97. These requirements do not conflict with current regulatory guidelines and are acceptable. However, they do not provide sufficient information to make a determination that the plant-specific design and arrangement will be adequate. Therefore, applicants referencing the passive Requirements Document must demonstrate compliance with the additional guidance in the SRP, or provide justification for alternative means of implementing the associated regulatory requirements.

5 SITE SECURITY SYSTEM

5.1 Scope and Functions

Section 5 of Chapter 9 of the passive Requirements Document specifies the requirements for a physical security system that will have the capability of protecting the facility against radiological sabotage, as required by 10 CFR 73.55. To achieve this function, Section 5.1.2 of Chapter 9 states that the site security system will be capable of, but not necessarily be limited to, (1) preventing unauthorized access of persons, vehicles, and materials to vital areas or protected areas; (2) detecting attempts by unauthorized persons to gain access to or to introduce unauthorized material across protected or vital area boundaries; (3) facilitating authorized activities and conditions within protected and vital areas; and (4) providing for authorized access to, while ensuring the detection assessment, and nullification of unauthorized penetrations of, protected or vital area boundaries.

The staff reviewed Section 5 of Chapter 9 for conformance to the regulations or regulatory guidance in 10 CFR 73.55; Section 13.6, Revision 2, "Physical Security," of the SRP (NUREG-0800); and NUREG-0908, "Acceptance Criteria for the Evaluation of Nuclear Power Reactor Security Plans." In addition, the staff considered NUREG-1267, "Technical Resolution of Generic Safety Issue A-29," and the Commission's "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," dated August 8, 1985, which stated: "The issues of both insider and outsider sabotage threats will be carefully analyzed and, to the extent practicable, will be emphasized in the design and in the operating procedures developed for new plants."

Section 5.1.3 of Chapter 9 provides design bases for the design of the site security system. In response to staff comments of May 17, 1991, EPRI amended the introduction to this section to clarify that these design bases do not supplant the requirements of 10 CFR 73.55(a), which references the design-basis threat assumptions of 10 CFR 73.1(a)(1). The design-basis assumptions and criteria specified in Section 5.1.3 of Chapter 9 are the following:

- The insider sabotage threat is based on one knowledgeable individual without armament or explosives.
- The security detection systems cannot be disabled without detection and timely response by the security force.
- Sabotage can potentially result in an initiating event requiring actuation of safe shutdown systems, in the disabling of safe shutdown systems, in the disabling of non-safety-related systems, or any combination of these.
- In evaluating the vulnerability to insider sabotage, it is assumed that outsiders cannot succeed in penetrating the protected area without detection.

- Equipment inside the containment is inaccessible to a saboteur during operation at power if security systems are provided to protect and control access to the containment.

- The continuous presence of several employees precludes acts of sabotage in the control room.

- Systems and equipment designated as vital for full-power operation will be maintained as vital during other modes of operation, except that during periods of plant shutdown, a vital area can be devitalized if approved in the physical security plan.

- Acts of sabotage are not assumed to occur concurrent with an independent single failure or other independently initiated event.

- The quality assurance program requirements of 10 CFR Part 50, Appendix B, do not apply to the security system, except at interfaces with safety systems where necessary to preclude adverse system interactions.

- Security restrictions for access to equipment and plant regions must be compatible with the need for access and exit for fire protection, with health physics access restrictions, and with local operator actions required for event mitigation. Security controls must provide for access during a loss of security system power and should not impede operator actions.

Initially, the staff did not accept EPRI's assumption that the presence of several persons in the control room precludes acts of sabotage absent a requirement ensuring that control room operators are aware of any maintenance activities or tampering with back panels that are out of their view. However, in its letter of July 22, 1991, EPRI stated that Chapter 10 of the passive Requirements Document includes appropriate requirements to justify the assumption. The staff reviewed Section 3.7.7.6 of Chapter 10 and noted a requirement for indication when testing is in progress that affects equipment operability. Therefore, it could be impossible for an individual to tamper with back panels without an operator becoming aware of it. This provision adequately addresses the staff's concern.

The staff will evaluate EPRI's assumption that a security area can be devitalized during unit shutdown, if approved in the physical security plan, during its review of an individual application for FDA/DC.

5.2 Performance Requirements

5.2.1 Protection Strategies

Section 5.2.1 of Chapter 9 of the passive Requirements Document requires that a physical security system be provided for protecting vital equipment in accordance with 10 CFR 73.55. Vital equipment requiring protection against radiological sabotage is defined in this section as (1) the reactor coolant boundary out to and including a single protected normally closed isolation valve; (2) the spent fuel pool and associated piping, equipment, and controls whose failure could result in an offsite release in excess of 10 CFR Part 100 limits; and (3) the equipment necessary for preventing core damage (including

redundant divisions). In the rationale for this section, EPRI identifies NRC Review Guideline 17 as the basis for determining which equipment needs to be designated vital. In this review guideline, RG 1.29, "Seismic Design Classification," is used as a reference for determining which equipment should be protected as vital equipment. In its response of May 13, 1991, to a request for additional information on Chapter 1 of the passive Requirements Document, EPRI confirmed that it considers that this linkage between equipment designated as seismic Category I and equipment designated as vital, in general, continues to be appropriate for the passive ALWR designs.

In its responses of July 22 and October 9, 1991, to staff questions on Chapter 5 of the passive Requirements Document, EPRI stated that the passive PWR containment steel shell and cooling air intakes and discharges are considered vital. The plant designer will analyze openings, or ports, in the concrete shield building needed for natural circulation of air for susceptibility to sabotage in a manner similar to that for openings through vital barriers for air intakes of evolutionary plant safety-related HVAC systems. EPRI also stated that the passive decay heat removal (PDHR) pool, condensing heat exchanger, and associated piping of the passive BWR are considered vital.

Chapter 3 of the passive Requirements Document requires redundant non-safety-related auxiliary systems to be capable of maintaining reactor coolant inventory and removing decay heat during transients and small loss of coolant accidents without reliance on the vital safety systems (except for the reactor protection system). In its letters of July 22 and October 9, 1991, EPRI stated that it does not consider these non-safety-related auxiliary systems to be vital equipment.

Section 5.2.1.1 of Chapter 9 includes as vital equipment the reactor coolant pressure boundary out to and including a single protected normally closed isolation valve. In its letter of April 17, 1991, the staff requested EPRI to clarify whether any components that could be used to open the protected normally closed isolation valves should be located in vital areas. The staff specifically asked EPRI to address whether local control stations and motor control centers from which the isolation valves for the following functions could be opened, or be prevented from closing, may need to be considered vital equipment:

- PWR reactor shutdown cooling (RSDC) system suction from the reactor coolant system hot leg
- BWR reactor water cleanup (RWCU) system suction from the reactor pressure vessel
- BWR main steamline drain and vent subsystem

In its response of October 9, 1991, EPRI stated that the pressure boundary isolation function for the passive plant is analogous to that for existing plants and that the protection for pressure boundary isolation equipment will meet 10 CFR 73.55 and be consistent with current practice. The staff does not consider this response to be sufficiently definitive to resolve its concern. It concludes that the level of design detail required to resolve this issue is beyond the scope of the passive Requirements Document. Resolution of this

issue can be deferred until review of an individual application for FDA/DC and until review of the completed vulnerability analysis and the vital equipment evaluation required by Sections 5.2.2 and 5.2.3 of Chapter 9 of the passive Requirements Document.

As discussed in Chapter 6 of this report, the staff considers that containing the non-safety-related auxiliary systems and their support systems behind at least one alarmed barrier (e.g., within the plant's protected area) will assist in providing additional defense-in-depth protection against external sabotage. In its letter of July 22, 1991, EPRI deleted the provision in Chapter 9 that explicitly required that the service water pumps be located within the protected area boundary where possible. In that letter, EPRI stated that the service water system will not be required to be located within a protected area. Nevertheless, Chapter 8 of the passive Requirements Document specifies that the service water system supply will be the basin of the circulating water cooling tower, and Appendices B and C of Chapter 6 show that the circulating water cooling towers for the BWR and PWR will be located within the plant's protected area. The staff considers that locating the service water system within the protected area (along with the onsite standby ac power supply and the non-safety-related auxiliary systems) may be important to ensure that a non-safety-related auxiliary system, such as the BWR RWCU system, is effective in providing additional defense-in-depth protection against external sabotage. However, as stated in Chapter 1 of this report, the staff concludes that evaluation of the effectiveness of the redundant non-safety-related auxiliary systems in providing additional defense-in-depth sabotage protection can be deferred until its review of a vendor- or utility-specific application and the sabotage vulnerability analysis that is required in a vendor- or utility-specific application by Section 5.2.2 of Chapter 9 of the passive Requirements Document. Furthermore, as discussed by the staff in Chapter 8 of this report, the staff and EPRI have agreed on a process to determine the regulatory treatment of non-safety-related systems for passive plant designs. Some systems, if they are classified as seismic Category I, may need to be protected as vital equipment.

Section 5.2.1 of Chapter 9 also requires that the physical security system be provided with tamper-detection alarms to ensure that the security force is cognizant of unauthorized disablement of the system.

5.2.2 Protection Methodology

Section 5.2.2 of Chapter 9 specifies that, before the plant design is completed, a sabotage vulnerability analysis will be conducted. The staff will evaluate the design during its review of an individual application for FDA/DC to verify that it includes (1) a vulnerability analysis that addresses insider tampering and (2) either a vulnerability analysis that addresses the external assault threat defined in 10 CFR 73.1(a)(1)(i) or requirements for such an analysis to be done on a site-specific basis.

5.2.3 Vital Equipment Evaluation

Section 5.2.3 of Chapter 9 specifies that a comprehensive listing and evaluation of all vital equipment, including piping, water sources, and power supplies, will be prepared for use in ensuring that all support functions vital to vital systems operation or required to mitigate a 10 CFR Part 100

release are located within a vital area. EPRI notes that 10 CFR 73.21 requires this information, once formulated, to be controlled as unclassified sensitive safeguards information (SGI) using the procedures in 10 CFR 73.57 and NUREG-0794. Plant-specific sabotage scenarios or vulnerabilities in the physical protection system are considered SGI. However, the staff concludes that this material may not need to be controlled as SGI. One of the conditions necessary before "documents or other matter" must be designated SGI in accordance with 10 CFR 73.21(b)(1)(vii) is that the physical protection measures (other than any general regulatory requirement stated in 10 CFR 73.55) afforded for the equipment or area, as described in either a physical security plan, safeguards contingency plan, or a plant-specific safeguards analysis, must also be specifically described in the "documents or other matter."

The staff will review individual applications for FDA/DC to ensure that a comprehensive listing and evaluation of vital equipment are included.

5.2.4 Vital Component Layout

Section 5.2.4 of Chapter 9 specifies that the locations of vital equipment and vital area barriers will be selected to be compatible with the requirements pertaining to physical separation, missile protection, fire protection, flooding protection, and access for maintenance. Access will require passage through at least two physical barriers (i.e., protected area boundary and vital area boundary). The plant designer will evaluate the security system's impact on normal and emergency operating activities.

In its letter of June 24, 1991, EPRI committed to clarify that the penetration delay times provided by vital doors and barriers should be consistent with the capabilities of the site's intrusion detection system and the armed response team. Sandia report SAND87-1926, "Access Delay Technology Transfer Manual," is cited as the basis for barrier delay times. The staff considers the minimum penetration times in this reference to be appropriate for this purpose. In its letter of October 9, 1991, EPRI stated that assumptions made by the plant designer regarding the response times of the site's intrusion detection system and armed response team would become interface criteria for applicants that reference the certified plant design. The staff concludes that this position is acceptable.

In its letter of June 24, 1991, EPRI also committed to amend Chapter 9, Section 5.2.4.1, to state that the layout of vital area boundaries should not interfere with operators being able to reach and enter sufficient divisions of safety equipment within appropriate time periods to prevent or mitigate accidents. The staff has confirmed that a statement has been added to state that the passive Requirements Document does not require locked security doors between the divisions of vital components.

5.2.5 Physical Protection Measures

Section 5.2.5 of Chapter 9 identifies the physical features to be incorporated into the plant design to facilitate optimization of the security system. These features include the following:

- Vital area walls, floors, and ceilings will be constructed to the requirements of 10 CFR 73.2 to delay penetration of these areas so that the security force has enough time to respond to an act of sabotage.

- The central alarm station, the control room, and the protected area access control station will have bullet-resistant features.

- Utility port (piping, HVAC, electrical, etc.) openings in protected area boundaries and vital area boundaries will be minimized and will include provisions to ensure they do not compromise security.

- Walls and doors will be designed to accommodate security system hardware. All security hardware (with the exception of the entrance card reader) will be located on the vital or secure side of doors.

5.2.6 Strategy for Inoperable Vital Equipment

Section 5.2.6 of Chapter 9 originally required a strategy to minimize the effect that an out-of-service division of vital equipment would have on the security force's compensatory measures. This section was deleted in Revision 3.

5.2.7 Protected Areas and Boundaries

Section 5.2.7 of Chapter 9 requires that physical protection measures be considered in physical site planning. It cites the following examples of physical site planning related to security:

- Site support buildings such as warehouses and office buildings will be located to facilitate required searches for access to the protected area.

- Personnel, vehicle, and cargo access portals will be located to expedite the processing required for plant personnel and vehicle traffic.

- The topography of the protected area boundary will be as flat as reasonably possible, and excessive changes in direction will be avoided.

- The isolation zone dimensions will be large enough to accommodate the intrusion detection system technology selected.

- To the extent possible, bodies of water will not coincide with the protected area boundary.

- Closed circuit television, lighting, and detection equipment in the isolation zone will not interfere with observation of the isolation zone on both sides of the protected area barrier.

- Intrusion alarms will indicate intrusion attempts by zone identification at consoles in the central alarm station (CAS) and the secondary alarm station (SAS).

- The number and size of subterranean passages under the protected area boundary will be minimized.

- Security systems and equipment, including the CAS, power supplies, and communications equipment, will be located within the protected area.

In the DSER for Chapter 9, the staff identified alarm assessment coverage of the interior of the intrusion detection system as a confirmatory issue because EPRI had committed to revise Section 5.2.7.1 of Chapter 9 to require unobstructed coverage of the area interior to the detection equipment, including the protected area fence, in order to ensure adequate alarm assessment. The staff has verified that EPRI's Revision 3 of Section 5.2.7.1 satisfactorily addresses this issue by the addition of guidance on the need for assessing the area of penetration. Therefore, this confirmatory issue is closed.

5.2.8 Design Margins

Section 5.2.8 of Chapter 9 specifies that security facilities will be designed with features to facilitate the replacement and modification of security equipment. Examples of such features include (1) maintenance access areas behind CAS and SAS consoles, (2) perimeter cable trenches for protected area perimeter security devices, (3) trenches or raised floors for cable routing in security equipment rooms, and (4) sizing of subsystem capacities (e.g., that of the security power supply, security computer, HVAC, and maximum number of detection and assessment zones) with margin for future expansion.

5.2.9 Training Facilities

Section 5.2.9 of Chapter 9 specifies that the plant designer or utility should consider locating the security operator training facility within the protected area and locating the qualification training facilities within the owner-controlled area but outside the protected area.

5.2.10 Access Control

Section 5.2.10 of Chapter 9 identifies the design features for portals controlling vehicle and personnel access to vital and protected areas. Required features include (1) means for positive identification of authorized individuals requiring and allowed access to protected areas; (2) means to search for and detect firearms, incendiary devices, and explosives; (3) means to control the last access into the protected area from within a bullet-resistant structure that requires a permit/concur actuation before access is allowed; (4) means for positive identification of individuals requiring and allowed access to vital areas; (5) alarming and logging of access portals; and (6) interfacing of access control with health physics and fire protection requirements so that only one door control mechanism is required. The access control will also allow for rapid movement through vital area boundaries in emergencies.

5.2.11 Communications

Section 5.2.11 of Chapter 9 requires that a security communications system be provided. The system will provide (1) each onsite security officer, watchman, or armed response individual with the capability for continuous two-way voice communications with the CAS, SAS, or personnel access portal (PAP);

(2) communications capability between the CAS and SAS and local law enforcement agencies; and (3) dedicated telephone communications capability between the main control room and the CAS and SAS.

5.2.12 Power Source

Section 5.2.12 of Chapter 9 specifies that the security system will have a backup power source. The security power supply will be alarmed at the CAS and SAS to ensure its availability.

In Revision 3, EPRI revised Chapters 9 and 11 to state that the security equipment onsite secondary power supply will be located in a vital area. A separate, dedicated, uninterruptible backup power source located in a vital area, including a security diesel generator, uninterruptible power supply (UPS) cabinets, and batteries, is required. The staff concludes that this is compatible with the applicable regulatory requirements in 10 CFR 73.55(e)(1) and 73.55(f)(4).

The security UPS battery capacity will be sized to power security monitoring equipment, including intrusion detection, alarm assessment, communications, and security lighting, for at least 2 hours following loss of all ac power sources. The security diesel generator will be sized to power all security systems, including lighting, for at least 24 hours without refueling. The staff concludes that this provides reasonable guidance for meeting GL 87-08, which states that the duration of the security secondary power supply capacity should at least equal the station blackout criterion applicable to the plant.

5.2.13 Data Management

Section 5.2.13 of Chapter 9 specifies redundant on-line dedicated security central computers in communication with remote security equipment. The remote security equipment will have sufficient computer capacity to permit interim stand-alone operation should communications with the central computers be interrupted. Good human factor design principles are to be used. Additional applicable man-machine interface requirements, including requirements for protection against computer viruses, are contained in Chapter 10 of the passive Requirements Document.

5.3 Testing and Maintenance

Sections 5.3 and 5.4 of Chapter 9 of the passive Requirements Document require that a test and inspection program be developed for the physical security equipment, with provisions for self-testing of computer, microprocessor, and multiplexor subsystems and components, and for verifying operability following maintenance, in order to verify the operability of security systems and to comply with 10 CFR 73.55(g). General requirements for maintenance in Chapter 1 of the passive Requirements Document also apply to security system maintenance.

5.4 Conclusion

The staff concludes that none of the requirements in Section 5 of Chapter 9 would prevent compliance with existing NRC security requirements and that acceptable design requirements to facilitate compliance with 10 CFR 73.55 have

been specified. The staff will evaluate satisfactory compliance with these requirements during its review of an individual application for FDA/DC. This evaluation will not replace the site-specific review of security, contingency, and guard training plans required by 10 CFR 50.34 and 10 CFR Part 73.

6 DECONTAMINATION SYSTEM (FACILITIES)

Section 6 of Chapter 9 of the passive Requirements Document gives the requirements for the various decontamination systems and/or facilities that will be used to remove or reduce radioactive contaminants from plant equipment, protective clothing, and personnel. The functions of the various decontamination systems and/or facilities include the following:

- to reduce area radiation dose rates to an acceptable level for subsequent maintenance, inspection, or repair
- to reduce contamination and radiation levels to permit the disposal of equipment or to facilitate the repair of equipment, consistent with as-low-as-is-reasonably-achievable principles
- to remove as much surface activity as is required so as not to compromise subsequent plant operations
- to provide facilities and supplies for personnel decontamination

Decontamination systems (including electropolishing units and degreasing units) for plant equipment will be sized to accommodate large components that may require decontamination on a regular basis. Decontamination facilities will also be available for the decontamination of small tools and instruments. The specific decontamination techniques that will be used at each plant will be selected by the plant designer. However, EPRI recommends the use of the decontamination techniques that are specified in EPRI Reports NP-6433 and NP-2777.

Section 6.2.2 of Chapter 9 states that areas for decontaminating equipment and personnel will be provided in the primary containment, auxiliary and turbine buildings, health physics facilities, and contaminated shops. These areas will have provisions for temporary shielding.

Section 6.2.3 of Chapter 9 states that permanently established decontamination areas will be provided with alarmed radiation monitors to alert personnel to unexpected radiation levels near decontamination equipment.

Section 6.3.2 of Chapter 9 states that a clean staging, decontamination, and checkout area will be provided for applying and removing protective materials that are used for tools and instruments that will be used in the containment during outage inspection and maintenance activities. This area would reduce congestion at the containment entrances and exits and would expedite worker traffic into and out of the containment. An area will also be provided for storing contaminated equipment awaiting decontamination or disposal.

Section 6.2.4 of Chapter 9 states that the plant designer will evaluate potential decontamination methods and equipment to ensure that chemicals and radioactive waste generated during the decontamination processes are maintained within the plant radioactive waste disposal goals specified in

Chapter 12. Procedures must be in place to ensure that systems and components are properly isolated during decontamination and are flushed to remove decontaminants following decontamination. Means will be provided to transfer chemical and radioactive wastes safely from the systems or components being decontaminated to the appropriate radioactive waste processing systems.

RG 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Reasonably Achievable," states that radiation doses to personnel can be reduced by decontaminating systems or components before they are serviced. The staff concludes that the decontamination methods and facilities described in Section 6 of Chapter 9 of the passive Requirements Document are intended to facilitate decontamination of equipment, protective clothing, and personnel and comply with the guidelines in RG 8.8.

In addition, the staff concludes that the specifications for the decontamination system provided in Section 6 of Chapter 9 are consistent with the criteria of SRP Sections 11, "Radioactive Waste Management," and 12, "Radiation Protection," and are, therefore, acceptable.

7 COMPRESSED AIR AND GAS SYSTEMS

7.1 Compressed Air System

Section 7 of Chapter 9 of the passive Requirements Document specifies that the compressed air system will consist of three separate and isolated subsystems: the plant service air system, the instrument air system, and the breathing air system. The plant service air system will provide a continuous supply of dry compressed air for air-operated tools, miscellaneous equipment, and various maintenance purposes. The instrument air system will provide a continuous supply of dry, oil-free, filtered, compressed air to all air-operated instrumentation and valves in accordance with the guidelines in American National Standards Institute (ANSI) MC11.1-1976 (ISA-57.3), "Quality Standard for Instrument Air Systems." The breathing air system will supply clean, oil-free, low-pressure air to various locations in the auxiliary building and in the containment for protection against airborne contamination during certain maintenance, inspection, and cleaning operations. These systems will not be safety related, with the exception of containment penetrations, accumulators, and check valves upstream of accumulators used for air-operated safety-related valve actuators in other ALWR systems that are designated as safety related.

Section 7.2.1.3 of Chapter 9 states that the air and gas system piping between the outer containment isolation valves and the accumulators will also be safety related. EPRI states that these systems are not required to achieve safe reactor shutdown or to mitigate the consequences of an accident. Failure of the non-safety-related portions of the plant service air, instrument air, and breathing air systems will not prevent safety-related components or systems from performing their intended safety functions. The staff's evaluation of the safety-related portions (containment penetrations) of these systems is provided in Chapter 5 of this report.

Section 7.3.4 of Chapter 9 requires that applicants referencing the passive Requirements Document have a periodic instrument air quality sampling program and that sample lines and valves be provided for obtaining air samples from air-operated safety-related plant equipment. These requirements, which are not required for the instrument air systems used in current nuclear power plants, will enhance the reliability and availability of the ALWR instrument air system.

Section 7.3.3 of Chapter 9 specifies that the plant breathing air quality will meet current ANSI breathing air quality standards. Complete isolation of different air supplies, including the breathing air supply, will be ensured by using separate, all-welded air supply headers and piping. Inadvertent cross connection of different gas supply systems will be prevented by using unique air fittings and identification tags. RG 8.8 states that the use of respiratory protection, including the use of supplied breathing air, is acceptable when the application of engineering controls is not feasible for providing protection against airborne radioactive material. Therefore, the staff concludes that the design features described in this section, to ensure that quality breathing air is available for respiratory protection, are acceptable.

Several incidents (e.g., the cutting off of the air supply to vital instruments because excessive moisture in the instrument air line froze, the

presence of desiccant particles in the instrument air systems that contributed to the loss of the salt water cooling system at San Onofre, and the slow closure of a containment isolation valve at Rancho Seco) prompted the staff to issue GL 88-14, "Instrument Air Supply System Problems Affecting Safety-Related Equipment," dated August 8, 1988, to help ensure the reliability and availability of the instrument air systems in operating plants. Table B.1-2, "NRC Regulatory Guidance for Design of ALWR Passive Plant," in Appendix B to Chapter 1 of the passive Requirements Document incorporates the requirements of GL 88-14 that are applicable to instrument line designs. EPRI states that the ALWR instrument air system will be separated from the breathing air system and service air system. The instrument air system design criteria include (1) maintaining the water vapor dew point for outside lines at an operating pressure corresponding to 18 °F below lowest outdoor temperature, (2) maintaining the water vapor absolute dew point upper limit at 20 °F for indoor installation, and (3) limiting particulates to less than 1 micron. In addition, EPRI committed to incorporate requirements in the passive Requirements Document that the plant designer evaluate the instrument air system to ensure that check valves isolate in the event of a slow pressure loss in the air supply system and provide for such sampling capability to ensure continued operation of the check valves and accumulators. The staff concludes that EPRI has improved the ALWR instrument air system design to ensure that moisture in the instrument air line does not freeze and has incorporated the requirements of GL 88-14 into the passive Requirements Document to ensure the reliability of the instrument air system.

7.2 Compressed Gas System

Section 7 of Chapter 9 states that the function of the compressed gas system is to provide pressure-regulated supplies of various gases needed for cooling, purging, diluting, inerting, and welding. The compressed gas system will consist of the following subsystems: nitrogen, hydrogen, oxygen, carbon dioxide, argon/methane, acetylene, and argon systems. These systems will not be safety related, with the exception of containment penetrations that are designated as safety related. Failure of the compressed gas system will not prevent safety-related components or systems from performing their intended safety functions. The staff's evaluation of the safety-related portions (containment penetrations) of these systems is given in Chapter 5 of this report.

7.3 Conclusion

The staff concludes that the design requirements in Section 7 of Chapter 9 for the compressed air system do not conflict with the guidance in SRP Section 9.3.1, "Compressed Air System," or RG 8.8 and are, therefore, acceptable.

8 HEATING, VENTILATING, AND AIR CONDITIONING SYSTEM

In the DSER for Chapter 9, the staff stated that, during its review of Section 8 of Chapter 9 of the passive Requirements Document, it noted multiple examples where the EPRI requirements did not explicitly require compliance with the SRP, the GDC, regulatory guidance, or Commission regulations. In addition, in several instances, EPRI stated that HVAC systems and components would be designed for seismic loads in accordance with Uniform Building Code (UBC), Zone 2A. However, the staff concluded that the use of this code might not be adequate to provide the needed reliability. Finally, in several sections, EPRI did not specify power sources or adequately describe how the system or component would perform following a loss of offsite power. The above issues were summarized in Section 8.5 of the DSER for Chapter 9 and were identified as open issues. The disposition of those open issues is given in Section 8.5 of this report.

8.1 Scope and Functions

Section 8.1 of Chapter 9 of the passive Requirements Document states that the functions of the HVAC system are as follows:

- to maintain work environments within the comfort levels required for operating and maintenance personnel
- to ensure the proper environment is maintained for equipment and structures
- to ensure that air flow is not directed from a volume with a higher potential for airborne radioactivity to one with a lower potential
- to ensure that plant areas are segregated in response to signals from the fire protection system
- to ensure that the HVAC design meets security system requirements for bullet resistance and personnel barriers
- to maintain appropriate relative pressure within a building or volume to ensure controlled leakage of potentially radioactive effluents
- to remove and inhibit the spread of contamination by providing appropriate HVAC filtration systems
- to ensure that ventilation system exhausts to the environment are within the limits specified in 10 CFR Part 20 for normal conditions
- to ensure that the design objectives of 10 CFR Part 50, Appendix I, can be met for normal operation and anticipated operational occurrences
- to prevent migration of smoke, hot gases, and fire suppressants into other fire areas, to the extent that safe shutdown capabilities, including operator actions, could be adversely affected

8.2 Performance Requirements

Section 8.2 of Chapter 9 of the passive Requirements Document establishes the key requirements for the design of HVAC systems applicable to both BWRs and PWRs.

In a letter dated August 1, 1991, EPRI responded to a staff request for information by stating that the only safety-related HVAC ductwork for passive plants would be the containment penetration isolation components, which would be classified as Safety Class 2, seismic Category I. These components are part of the containment pressure boundary and will be designed in accordance with ASME Code, Section III containment penetration requirements. The staff concluded that this response was acceptable.

Section 8.2.5 of Chapter 9 states that the pump house ventilation system will be a non-safety, non-seismic system. In the DSER for Chapter 9, the staff stated that this was inconsistent with the way EPRI had treated other HVAC systems that traditionally have been Safety Class 3, seismic Category I systems. The staff concluded that EPRI had not provided sufficient justification for this inconsistency.

EPRI states that all other HVAC systems in the passive ALWRs will be designed as non-safety systems. The general design requirements in Section 8.2.1 of Chapter 9 reflect this approach to HVAC design. This is in contrast to both the current generation of LWRs and the evolutionary ALWRs in which HVAC systems, used for mitigating the radiological consequences of an accident or for supporting safety systems, are safety grade. For those HVAC systems that traditionally have been safety grade, the passive Requirements Document specifies performance, system, and equipment design requirements. These include design for seismic loads (UBC Zone 2A), redundancy, the availability of non-safety-grade electric power, and protection against internal and external hazards. Although it does not provide specific requirements for the reliability of these systems, EPRI, in response to staff questions, indicated that it was evaluating specific reliability targets and other measures to provide confidence that the passive plants will meet performance requirements, and that it will address both passive safety and active non-safety systems. Although a system may not need to meet all of the safety-grade criteria, it may be appropriate to allow some credit for non-safety ventilation and filtration systems on the basis of reliability considerations. The extent of this credit will be determined as part of the staff's review of the regulatory treatment of non-safety systems (RTNSS).

In the DSER for Chapter 9, the staff identified the classification of HVAC systems (including the pump house ventilation system) as an open issue. In its letter dated September 29, 1992, EPRI stated that the passive plant HVAC systems are non-safety grade and non-seismic and that they would be designed in accordance with UBC seismic Zone 2A, as required by Section 4.3.2.3 of Chapter 1 of the passive Requirements Document. In addition, EPRI agreed with the staff position in the DSER that this open issue should be resolved as part of the policy issue associated with RTNSS. The regulatory treatment of non-safety HVAC systems will be in accordance with the resolution of Policy Issue III.A, "Regulatory Treatment of Non-Safety Systems for Passive ALWRS," which was identified in Annex A of Appendix B to Chapter 1 of the DSER. In SECY-94-084, the staff describes a process for resolving of Policy

Issue III.A. In this process, the ALWR designer will identify all risk-significant non-safety SSCs' functions and their corresponding reliability/availability (R/A) missions. Furthermore, the designer will propose regulatory treatment appropriate to provide confidence that the identified R/A mission will be achieved. The staff will evaluate the results of this process during its review of an individual application for FDA/DC to determine if enhanced regulatory oversight is required for these systems. Therefore, this open issue is closed.

8.2.1 Requirements for All HVAC Systems

The following is a summary of the key requirements for HVAC systems given in Section 8.2.1.1 of Chapter 9:

- HVAC systems will provide a suitable environment in accordance with the recommendations of the American Society of Heating, Refrigerating, and Air Conditioning Engineers to ensure the safety and comfort of plant personnel and the operability of plant equipment during normal operation.

- HVAC systems of areas or buildings that may contain radioactivity will be designed in accordance with 10 CFR Part 20; RG 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants"; American National Standards Institute/American Society of Mechanical Engineers (ANSI/ASME) N509 and N510; ANSI/American Nuclear Society 59.2; and ANSI/ASME PG-1-1988, Article AA-4000 to

  - maintain negative ambient pressures with respect to the atmosphere during normal operation

  - filter exhaust air to reduce controlled radioactivity releases to the environment

  - provide continuous exhaust air monitoring (with high-level alarm-equipped monitors in the control room) of each potentially contaminated discharge point

  - provide internal cleanup and/or an outside air supply for maintaining the levels of maximum permissible concentrations within as-low-as-is-reasonably achievable (ALARA) guidelines

  - provide means to locate and isolate the ventilation air of an area in which there is a radioactive leak.

- HVAC systems will be designed so that air flow is not directed from volumes with a higher potential for airborne radioactivity to volumes with a lower potential.

- Penetration openings for ventilation systems in fire-rated barriers will be protected by fire dampers having a rating equivalent to that required for the barrier.

- HVAC duct penetrations through security boundaries will be equipped with an appropriate barrier.

- HVAC systems will be provided with sufficient instrumentation and controls to ensure safe, efficient, and reliable operation.

- Instruments and instrument taps of proper size, sufficient quantity, and properly located will be provided to allow initial and periodic testing of components and equipment to demonstrate their operation within performance limits.

- All nuclear air treatment filtration systems will be designed, fabricated, installed, and tested in accordance with RG 1.140 and ANSI/ASME N509 and N510.

- The system heat removal capacity will include a margin of 15 percent of the total heat load and 15 percent of system pressure loss.

- HVAC systems will be designed to withstand seismic loads without affecting operability and will be capable of maintaining design conditions (flow, temperatures, pressures) while the largest capacity component (fan, filter, coil) is inoperable because of maintenance or repair.

- The plant designer will identify all outside air intake and exhaust openings to be provided with tornado protection. The level of tornado protection will be consistent with the design of the structure in which the identified openings are located.

- All HVAC systems and components will be designed or protected so that their failure will not cause interference or adverse interactions with the proper functioning of seismic Category I systems.

- If the plant designer determines that charcoal filters are necessary to meet the requirements of 10 CFR Part 20 and 10 CFR Part 50, Appendix I, and to maintain in-plant ALARA levels in accordance with RG 8.8, non-safety charcoal filters will be designed with the following features to process radioactivity from normal and off-normal operation:

  - charcoal filters and associated components designed in accordance with ANSI/ASME N509-1989 and RG 1.140

  - prefilter and high-efficiency particulate air (HEPA) filter upstream of charcoal absorber

  - electric heating coils powered from auxiliary power source

  - postfilter to retain carbon fines

- Controls for redundant HVAC trains will be integrated with the plant fire detection and protection plan to prevent smoke and hot gases from migrating between redundant HVAC trains.

- Motors, fans, dampers, and other components of HVAC systems will be designed to operate in environments associated with the normal conditions to which they will be exposed.

- HVAC systems will be designed to permit appropriate inservice inspections and functional testing.

- Isolation components of containment penetrations will be Safety Class 2, seismic Category I.

Section 8.2.1.1.25 of Chapter 9 states that HVAC systems and components will be designed or protected so that their failure will not cause interference or adverse interactions with the proper functioning of seismic Category I systems. In the rationale portion of this section, EPRI does not refer to "seismic Category - II/I criteria." However, the seismic Category II/I criteria still must be met for these systems. In the DSER for Chapter 9, the staff identified the classification of HVAC systems as an open issue. In its letter dated September 29, 1992, EPRI stated that the passive plant HVAC systems are non-safety grade and non-seismic and that they would be designed in accordance with UBC seismic Zone 2A, as required by Section 4.3.2.3 of Chapter 1 of the passive Requirements Document. Therefore, EPRI stated that this open issue should be resolved as part of the policy issue associated with RTNSS. The regulatory treatment of non-safety HVAC systems will be determined in accordance with the resolution of Policy Issue III.A, which was identified in Annex A of Appendix B to Chapter 1 of the DSER. In SECY-94-084, the staff describes a process for resolving of Policy Issue III.A. In this process, the ALWR designer will identify all risk significant non-safety SSCs' functions and their corresponding R/A missions. Furthermore, the designer will propose regulatory treatment appropriate to provide confidence that the identified R/A mission will be achieved. The staff will evaluate the results of this process during its review of an individual application for FDA/DC to determine if enhanced regulatory oversight is required for these systems. Therefore, this open issue is closed.

Section 8.2.1.1.26 of Chapter 9 originally required a postfilter instead of a HEPA filter downstream of the adsorber to collect carbon fines only if the discharge is to other safety-related equipment or occupied spaces. As discussed in Section 3.3.3 of Chapter 12 of this report, the staff believes that a simple postfilter will not be as effective as a HEPA filter in collecting carbon fines. Therefore, in the DSER for Chapter 9, the staff concluded that EPRI had not justified its position for assuming the RG 1.140 efficiencies for removing elemental and organic iodine from the effluent stream if there is no HEPA filter downstream of the charcoal adsorber for collecting carbon fines, as specified in the regulatory guide. This was identified as an open issue. In its letter dated January 26, 1993, EPRI amended Chapter 9, Section 8.2.1.1.26 and its rationale to resolve the open issue. The amended section also requires the filter system components to be designed in accordance with ANSI/ASME N509-1989 and RG 1.140. Specifically, it mandates the provision of a postfilter downstream of the charcoal adsorber, irrespective of where the discharge would go. Further, the amended rationale states explicitly that the efficiencies in Section 5.3.1 of ANSI/ASME N509-1989 for prefilters and postfilters are applicable. The staff notes that the subject section of the standard requires 95-percent efficiency for the postfilter downstream of the charcoal adsorber. The staff finds that the amended section and its rationale ensure that the carbon fines will be collected in a satisfactory manner and are, therefore, acceptable. Therefore, this open issue is closed.

EPRI indicates that the HVAC system will be designed to minimize the potential exposure of personnel to radioactivity from airborne contamination. To minimize time spent by personnel in radiation areas and to facilitate decontamination, the passive Requirements Document specifies that suitable facilities will be located to support maintenance and repair work performed in high-radiation areas and to support decontamination activities. These features to minimize personnel radiation exposure and to prevent the spread of radioactive contamination comply with the guidelines of RG 8.8 and are, therefore, acceptable.

The staff concludes that the general design requirements in Section 8.2.1 of Chapter 9 are commensurate with the non-safety classification of the HVAC systems and are similar to the guidance in SRP Section 9.4.1, "Control Room Area Ventilation System"; SRP Section 9.4.2, "Spent Fuel Pool Area Ventilation System"; SRP Section 9.4.3, "Auxiliary and Radwaste Area Ventilation System"; SRP Section 9.4.4, "Turbine Area Ventilation System"; and SRP Section 9.4.5, "Engineered Safety Feature Ventilation System."

8.2.2 Control Complex HVAC Systems

Section 8.2.2 of Chapter 9 states that the control complex will include the main control room envelope, the computer room, essential switchgear rooms, battery rooms, and HVAC equipment room. All of the control complex HVAC systems will be non-safety. They will be designed for seismic loads in accordance with the UBC, Zone 2A and will be capable of accomplishing their intended functions, assuming a single active component failure and a loss of offsite power. It is the staff's position that the control complex HVAC systems should be designed, at a minimum, to the provisions of UBC Zone 3 with a restriction that the duel systems and concentric braced frames not be used. As an alternative, the provisions of UBC Zone 2A may also be used for the control complex HVAC system design, provided the plant designer demonstrates that the control complex HVAC systems will retain its structural integrity under the specified safe shutdown earthquake (SSE). The staff will review individual applications for FDA/DC to determine the acceptability of the plant designer's demonstration of the structural integrity of the control complex HVAC systems under the specified SSE.

The control room envelope HVAC system will service the control room and the facilities provided for the convenience of the operators (i.e., shift supervisor's office, operator washroom, kitchen, etc.). Section 8.2.2.1 of Chapter 9 states that the system will have adequate redundancy. Redundant components will include air handling units (AHUs), smoke purge fans, supplementary filter units, associated power supplies, and controls. Each AHU will consist of a prefilter, a high-efficiency filter, a heating and cooling coil, and a supply fan. The cooling coils will be served by the chilled water system. Each supplementary filter unit will consist of a prefilter, a high-efficiency particulate air filter, and a supply fan.

EPRI states that all main control room HVAC system essential components will be designed for seismic loads in accordance with UBC, Zone 2A and will be capable of accomplishing their intended functions assuming a single active component failure and a loss of offsite power.

Section 8.2.2.1 of Chapter 9 states that all redundant equipment of the main control room HVAC system will be physically separated all and designed not to generate missiles. During normal operation, the main control room HVAC system will continuously maintain a slight positive pressure in the control room emergency zone.

Section 8.2.2.1 of Chapter 9 states that the control room envelope atmosphere will normally be maintained at a temperature ranging from 73 °F to 78 °F, with a 1-hour maximum of 85 °F, and at a relative humidity ranging from 25 to 60 percent during all operating modes. If all ac power in lost, HVAC dampers will isolate the control room and provisions will be made to limit the average room temperature rise to 15 °F maximum at the end of the postulated 72-hour accident.

Section 8.2.2.1 of Chapter 9 states that redundant detectors (radiation, toxic gas, and smoke) will be provided in the outside air intakes to isolate the main control room normal outside air isolation dampers on receipt of a high-level detection signal. The outside air dampers will be in the closed position before the detected contaminated air reaches the isolation damper's downstream side. The isolation dampers will be of leaktight construction with fail-close-type operators. When airborne radioactivity is high, outside makeup air will be supplied through the supplementary filter units if ac power is available and the HVAC system is operational. When high levels of toxic gas or outside smoke are detected, the control room envelope HVAC system will operate in the recirculating mode without a supply of outside makeup air.

In Section 8.2.2.1 of Chapter 9, EPRI also states that the control room envelope HVAC system will be capable of removing smoke from the control room after a fire upon a manual initiation; the quantity of outside air will be increased to 100 percent of the system's supply air flow rate. The redundant outside air isolation dampers at the control complex boundary will be leak-tight and seismic Category I.

Section 8.2.2.1 of Chapter 9 includes a requirement that exterior HVAC openings penetrating the main control room vital boundary be bullet resistant, for compliance with the requirements of 10 CFR Part 73.

In the DSER for Chapter 9, the staff noted that EPRI did not include a requirement to meet GDC 19 of Appendix A to 10 CFR Part 50 and concluded that EPRI's requirements for thyroid and beta skin radiation dose (with no breathing apparatus and protective clothing) had not been adequately justified. This was identified as an open issue. In Revision 4, EPRI revised its position and now requires that a passive control room pressurization system be provided so that the dose requirements in GDC 19 and SRP Section 6.4 will be satisfied. The staff's evaluation of the acceptability of EPRI's requirements for control room habitability is given in Section 6.5 of Chapter 5 of this report. Therefore, on the basis of that evaluation, this open issue is closed.

In addition, EPRI requires that the control room be designed so that it can be maintained during the 72-hour design-basis period as the primary location from which personnel can safely operate in case of an assumed accident and other postulated design conditions. The passive Requirements Document states that the control room ventilation system for passive plants will be designed with

appropriate redundancy and filtration (including charcoal filters) and will be powered both from onsite and offsite ac power sources. When ac power is available, control room habitability will be provided by the non-safety-grade HVAC system. However, EPRI states that ALWR designs should not rely on the HVAC system to meet the current radiation dose criteria in SRP Section 6.4. However, depending on the accident sequences, the duration of the accident can be much longer than 72 hours. GDC 19 states that "adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions...for the duration of the accident." Therefore, the staff concluded in Section 6.5 of the DSER for Chapter 5 that the duration of the accident, not just the EPRI-proposed 72-hour design-basis period, should be considered in the analyses of control room habitability. This was identified as an open issue. In Revision 4, EPRI revised its position and now requires that a passive control room pressurization system be provided so that the dose requirements in GDC 19 and SRP Section 6.4 will be satisfied. Also, safety-grade connections are required for the pressurization system to allow the use of offsite, portable air supplies, if needed after 72 hours to minimize operator doses. The staff's evaluation of the acceptability of EPRI's requirements for control room habitability is given in Section 6.5 of Chapter 5 of this report. Therefore, on the basis of that evaluation, this open issue is closed.

The staff concludes that the design requirements in Section 8.2.2.1 of Chapter 9 for the main control room HVAC system are consistent with the non-safety classification of the HVAC systems and are similar to the guidance in SRP Sections 6.4 and 9.4.1 and are, therefore, acceptable.

The staff's evaluation of the HVAC systems for the other areas of the control building is given in Section 8.2.5 of this chapter.

8.2.3 Onsite Standby AC Power Supply HVAC System

Section 8.2.3.1 of Chapter 9 establishes the requirements for the design of the diesel onsite standby ac power supply building ventilation system, which will consist of a normal and a supplementary ventilation subsystem for each generator compartment. Onsite ac power may be supplied by diesel generators and/or a combustion turbine. The key requirements include the following:

- The normal ventilation subsystem will be designed as a non-safety and non-seismic system.

- The normal ventilation subsystem will consist of a manually started, 100-percent-capacity fan; fresh air intake and modulated recirculating air damper; associated ductwork; and unit heaters.

- The standby ac power supply building will contain a separate exhaust system or vent piping for the combustible vapor from the fuel oil day tank.

- The supplementary ventilation subsystem will be designed to withstand earthquakes in accordance with UBC, Zone 2A and be capable of accomplishing its intended function assuming a single active failure and loss of offsite power.

- The supplementary ventilation subsystem will be designed to maintain the room temperature in the range given in Section 8.2.1.1.1 of Chapter 9 when the diesel generator is operating. The design will include local heating or cooling of enclosed spaces/panels, as necessary, to maintain the environment of the spaces within the allowable temperatures for electronic/control equipment.

- The supplementary ventilation subsystem will consist of fan(s); outside air and modulating return air dampers; cooling coil(s) supplied by chilled water, if necessary; and associated ductwork.

- The supplementary ventilation subsystem fan will be started automatically whenever the generator is started. The fan will stop automatically on low temperature. The subsystem will trip and isolate on fire detection. Provisions will be made to remove smoke after a fire. Also, the fan will be equipped with a manual start for testing and maintenance.

- The supplementary ventilation subsystem will be controlled from the control room.

Section 8.2.3 of Chapter 9 states that the supplementary ventilation subsystem should be designed to withstand earthquakes in accordance with UBC, Zone 2A.

Section 8.2.3 of Chapter 9 states that the supplementary ventilation subsystem will be capable of accomplishing its intended function, assuming a single active component failure and loss of offsite power. However, the passive Requirements Document does not specify the power sources for the onsite standby ac power supply to the HVAC system and how the system will perform following a loss of offsite power.

The staff concludes that the design requirements in Section 8.2.3.1 of Chapter 9 for the onsite standby ac power supply building ventilation system are commensurate with the system's non-safety classification and are similar to the guidance in SRP Section 9.4.5 and are, therefore, acceptable.

8.2.4 Fuel Handling/Spent Fuel Pool Area Heating and Ventilating System

Sections 8.3.3.1.5 and 8.4.4 of Chapter 9 establish the requirements for the design of the fuel handling/spent fuel pool area heating and ventilating system. The functions of the system include the following:

- to maintain the air temperature and humidity in the fuel handling area within an acceptable range so that plant personnel can perform their required tasks, and undue environmentally induced degradation of equipment and structures is avoided

- to maintain negative air pressure in the fuel handling area to preclude uncontrolled release of airborne radioactivity to the environment under normal operating conditions and in the event of a fuel handling accident

- to limit the airborne radioactivity levels in the fuel handling area to acceptable levels

In BWRs, on receipt of a high-radiation signal on the refueling floor, the system will change its operating mode as follows:

- trip all running ventilation fans and prevent redundant fans from starting or operating
- close the normally open isolation dampers for isolation of the secondary containment
- start the reactor building filtered exhaust system (the staff's evaluation is given in Section 8.3.4 of this chapter)

For PWRs, Section 8.4.4 of Chapter 9 states that the fuel facility ventilation system will be designed as a "once-through" system consisting of an exhaust and a supply subsystem. The key requirements include the following:

- The exhaust subsystem will be capable of automatic transfer from its normal (filter) bypass operating mode to a secondary filtration mode on detection of high radiation in the exhaust duct (i.e., fuel handling accident).
- The exhaust subsystem will be controlled from the control room. A local panel will also be provided with fan status lights and alarms.
- The supply subsystem will be designed as a non-safety, non-seismic subsystem supplied by the permanent non-safety distribution system and will be functional during normal plant operation only.
- The exhaust subsystem will be designed to withstand earthquakes in accordance with UBC, Zone 2A and be capable of accomplishing its intended function assuming a single active component failure and loss of offsite power. It will be provided with redundant systems, each consisting of a prefilter, a high-efficiency particulate air filter, an exhaust fan, a booster fan, and associated ductwork, controls, and instrumentation.

The passive Requirements Document does not require the provision of charcoal filters for the fuel facility ventilation system. However, in Section 8.2.1.1.26 of Chapter 9, EPRI states that the plant designer will perform analyses and evaluations to determine if charcoal filters are necessary to meet the requirements of 10 CFR Part 20 and 10 CFR Part 50, Appendix I, and to maintain in-plant ALARA levels in accordance with RG 8.8. The staff will evaluate the detailed system design operation and layout against the applicable guidelines of SRP Section 9.4.2, "Spent Fuel Rod Ventilation System," during its review of an individual application for FDA/DC. Additionally, the staff will review the need for charcoal filters for the system against the requirements of 10 CFR Part 20 and 10 CFR Part 50, Appendix I.

EPRI also states that after a fuel handling accident, only a small amount of air will be exhausted from the fuel facility in order to maintain negative air pressure within the fuel facility to preclude uncontrolled releases of airborne radioactivity to the environment. The amount of makeup air required will be drawn in through some predetermined openings. Therefore, the supply subsystem will not be needed and will not be designed as safety related. In the DSER for Chapter 9, the staff stated that because of the flexibility in

the system design, it would evaluate the detailed system design and operation and facility layout during its review of an individual application for FDA/DC. In its letter dated September 29, 1992, EPRI stated that the passive plant HVAC systems are non-safety grade and are not required to perform any safety-related functions. Therefore, EPRI stated that this item should be resolved as part of the policy issue pertaining to RTNSS. The regulatory treatment of non-safety HVAC systems will be determined in accordance with the resolution of Policy Issue III.A, which was identified in Annex A of Appendix B to Chapter 1 of the DSER. In SECY-94-084, provided as Annex B of this appendix, the staff describes a process for resolving Policy Issue III.A. In this process, the ALWR designer will identify all risk-significant non-safety SSC functions and their corresponding R/A missions. Furthermore, the designer will propose regulatory treatment appropriate to provide confidence that the identified R/A mission will be achieved. In its SRM dated June 30, 1994, the Commission approved the staff's recommendation on RTNSS. The staff will review the results of this process during its review of an individual application for FDA/DC to determine if enhanced regulatory oversight is required for this system.

Section 8.4.4 of Chapter 9 states that the exhaust subsystem will be designed to withstand earthquakes in accordance with UBC, Zone 2A. The provisions of UBC Zone 2A may be used for the exhaust subsystem design, provided the plant designer demonstrates that this subsystem will retain its structural integrity under the specified SSE. The staff will review individual applications for FDA/DC to determine the acceptability of the plant designer's demonstration of the structural integrity of the exhaust subsystem under the specified SSE. Section 8.4.4 of Chapter 9 states that the system will be capable of accomplishing its intended function assuming a single active failure and loss of offsite power. However, the passive Requirements Document does not specify the power sources for the exhaust subsystem or how the system will perform following a loss of offsite power.

The staff concludes that the design requirements in Chapter 9 for the fuel handling/spent fuel pool area heating and ventilating system do not conflict with the guidance in SRP Section 9.4.2 and are, therefore, acceptable.

8.2.5 HVAC Systems for Miscellaneous Areas

In various sections of Chapter 9, EPRI establishes specific requirements for the design of the HVAC systems for miscellaneous areas (e.g., essential switchgear rooms, battery rooms, and service water pump house; computer room; safety-related HVAC equipment room; combustion turbine building; radwaste facility; turbine building; radiological access control building; hot machine shop; clean shops; technical support center).

Sections 8.2.2.3 and 8.2.2.4 of Chapter 9 do not require that the ventilation systems supplying the Class 1E switchgear and battery rooms meet the associated requirements of GDC 2, 3, and 4 and the guidance in SRP Section 9.4.5.

Section 8.2.2.4.1 of Chapter 9 gives the design requirements for the hydrogen concentration (and the method for measurement) allowable in the essential battery rooms. The 2-percent hydrogen limit specified by EPRI is one-half of the 4-percent lower flammability limit and is in accordance with the provision

of IEEE 484. Because the 2-percent concentration is intended to be a HVAC design requirement and not an instrumentation requirement, it does not preclude HVAC instrumentation and is, therefore, acceptable.

Section 8.2.6 of Chapter 9 does not require that the radiological facility control building ventilation system meet the requirements of 10 CFR Part 20.

Section 8.2.7 of Chapter 9 does not require that the radwaste facility ventilation system meet the requirements of GDC 4 and the guidance in SRP Section 9.4.3.

In Revision 4, EPRI revised Section 8.2.2.4 Chapter 9 require that designers perform a thermal analysis of the Class 1E battery room HVAC system and associated equipment spaces to establish the normal operating temperature ranges that will ensure that the equipment in these spaces will not exceed allowable limits for 72 hours following a design-basis accident.

Section 8.2.9 of Chapter 9 does not require that the hot machine shop ventilation system meet the guidance in SRP Section 9.4.3.

In the DSER for Chapter 9, the staff stated that Section 8.2.12 of Chapter 9 did not discuss fans for the ventilation system for warehouse areas storing Level A items. This omission was identified as an open issue. In its letter dated September 29, 1992, EPRI stated that Section 8.2.12.1.1 of Chapter 9 describes the design temperature ranges and relative humidity requirements for Level A storage warehouse(s). The omission of fans from the list of necessary equipment for warehouse areas was an oversight and fans are needed to make the other components functional and to meet the temperature criteria specified. In its letter dated July 9, 1993, EPRI revised Section 8.2.12.1.1 by adding fans to the list of necessary equipment for warehouse areas. The staff concludes that EPRI's response and this revision clarify the requirements of Section 8.2.12.1.1. Therefore, this open issue is closed.

Section 8.2.15 of Chapter 9 does not require that the technical support center ventilation system meet the requirements of GDC 19 in accordance with NUREG-0737, Item II.B.2.

Sections 8.3.5 and 8.4.5 of Chapter 9 do not require that the turbine building ventilation systems for BWRs and PWRs meet the guidance of SRP Section 9.4.3.

The staff concludes that, except as noted above, the design requirements in the passive Requirements Document for the HVAC systems for these miscellaneous areas do not conflict with the guidance of SRP Sections 9.4.3, 9.4.4, and 9.4.5. However, the details (i.e., design layout, system performance, etc.) are insufficient for the staff to determine if these HVAC systems will be properly designed. Therefore, the staff will evaluate these HVAC systems against the above-cited SRP sections during its review of an individual application for FDA/DC. In addition, the staff will evaluate the need for charcoal filters for these systems on the basis of 10 CFR Part 50, Appendix I.

8.3 Performance Requirements and Interfaces for BWRs

8.3.1 BWR Reactor Containment Ventilation System

Section 8.3.2.1 of Chapter 9 of the passive Requirements Document establishes the design requirements for the BWR reactor containment ventilation system. The key requirements include the following:

- The system will be designed as a non-safety, non-seismic system powered from the permanent non-safety distribution system.
- During normal operation, the system will maintain temperatures in accordance with ANSI/ANS 56.7 and the manufacturer's recommendations.
- During cold shutdown or refueling, the system will provide filtered, heated or cooled outside air for ventilating or purging the containment to maintain ambient temperatures within the range given in Section 8.2.1.1.1 of Chapter 9 of the passive Requirements Document.
- The system will consist of properly located fan coil units that are adequately redundant; that is, the system will be fully operational with one fan coil unit out of service.
- Fan coil units that are required to operate during loss of offsite power will be powered from the permanent non-safety distribution system on loss of offsite power.

The staff concludes that the design requirements in Section 8.3.2.1 of Chapter 9 for the BWR reactor containment ventilation system do not conflict with the guidance in SRP Sections 9.4.3 and 9.4.5 and are, therefore, acceptable.

8.3.2 BWR Reactor Building Ventilation System

Section 8.3.3 of Chapter 9 establishes the requirements for the design of the BWR reactor building ventilation system. The key requirements include the following:

- The system will be designed as a non-safety, non-seismic system equipped with safety-related, seismically qualified, redundant isolation dampers. The system will be powered from the permanent non-safety distribution system and will operate during normal plant conditions only.
- The system will be designed to maintain the general area ambient temperatures within the ranges given in Section 8.2.1.1.1 of Chapter 9 of the passive Requirements Document; to provide ventilation, cooling, and heating to all equipment in the reactor building, including the engineered safety feature compartments, during normal plant operation; and to provide filtered heated or cooled outside air to maintain the maximum permissible concentration levels in the cubicles within ALARA guidelines.

Section 8.3.3 of Chapter 9 does not require that the reactor building ventilation system meet the guidance of SRP Section 9.4.3 and the requirements of GDC 60, 10 CFR 50.36a, Appendix I to 10 CFR Part 50, and 10 CFR Part 100.

Section 8.3.3.1.1 requires that the BWR reactor building ventilation system maintain ambient pressure at a minimum negative pressure of approximately 0.25-inch water gauge with respect to all adjacent spaces.

The staff concludes that, except as noted above, the design requirements in Section 8.3.3 of Chapter 9 for the BWR reactor building ventilation system do not conflict with the guidance in SRP Sections 9.4.3 and 9.4.5 and, therefore, are acceptable.

8.3.3 BWR Reactor Building Filtered Exhaust System

Section 8.3.4 establishes the requirements for the design of the reactor building filtered exhaust system, which will be required to perform the following functions:

- Exhaust the required quantity of reactor building air to maintain a slight negative ambient pressure during high radiation conditions.
- Filter the exhausted air before it is discharged.
- Filter the reactor containment purge exhaust air, if required.

In Revisions 2 and 4, EPRI revised Section 8.3.4 to remove the requirement that this system function during accident conditions. Instead, it will be available during normal plant operation to reduce undesirable levels of airborne contamination. The system will be designed for manual start from the main control room and controlled from a main control room panel.

The staff concludes that the design requirements in Section 8.3.4 of Chapter 9 for the reactor building filtered exhaust system do not conflict with the guidance in SRP Section 9.4.5 and are, therefore, acceptable.

8.4 Performance Requirements and Interfaces for PWRs

8.4.1 PWR Containment Cooling and Ventilation System

The PWR containment cooling and ventilation system will consist of five subsystems that will maintain environmental conditions within the containment during normal operating conditions, extended shutdowns, and refueling outages. These subsystems, which will not be required to perform any safety-related function, are the containment fan cooler subsystem, the containment cleanup subsystem, the control rod drive mechanism ventilation subsystem, the reactor cavity cooling subsystem, and the primary containment purge subsystem.

Section 8.4.2 of Chapter 9 of the passive Requirements Document establishes the requirements for the design of these subsystems. The key requirements for each of these subsystems are the following:

Containment Fan Cooler Subsystem (Section 8.4.2.1 of Chapter 9)

- The subsystem will be designed as a non-safety and non-seismic installation that will maintain area temperatures below 120 °F.

- The subsystem will consist of properly located fan cooler units that are adequately redundant; that is, the subsystem will be fully operational with one fan coil unit out of service.

- The fan coolers will be manually started from the control room. If the operating unit should fail, the standby unit will start automatically and an alarm will be annunciated in the control room.

- System operation will be monitored in the control room by fan cooler discharge air and water temperature indicators and high-temperature alarms.

- Fan coolers that are required to operate during loss of offsite power will be powered from the permanent non-safety power distribution system on loss of offsite power.

- Depending on the nuclear steam supply system design, this subsystem may also provide cooling to the reactor cavity area.

In the DSER for Chapter 9, the staff stated that in Section 8.4.2.1 of Chapter 9, EPRI did not require the system to meet "Seismic II/I criteria." This was identified as an open issue. In the DSER, the staff also identified the safety classification of HVAC systems as an open issue. In its letter dated September 29, 1992, EPRI stated that the passive plant HVAC systems are non-safety grade and non-seismic, and that they would be designed in accordance with UBC seismic Zone 2A, as required by Section 4.3.2.3 of Chapter 1 of the passive Requirements Document. Therefore, EPRI stated that these open issues should be resolved as part of the policy issue associated with RTNSS. The regulatory treatment of non-safety HVAC systems will be determined in accordance with the resolution of Policy Issue III.A, which was identified in Annex A of Appendix B to Chapter 1 of the DSER. In SECY-94-084, provided as Annex B of this appendix, the staff describes a process for resolving Policy Issue III.A. In this process, the ALWR designer will identify all risk significant non-safety SSC functions and their corresponding R/A missions. Furthermore, the designer will propose regulatory treatment appropriate to provide confidence that the identified R/A mission will be achieved. In its SRM dated June 30, 1994, the Commission approved the staff's recommendations on RTNSS. The staff will evaluate the results of this process during its review of an individual application for FDA/DC to determine if enhanced regulatory oversight is required for this system. Therefore, these open issues are closed.

Containment Cleanup Subsystem (Section 8.4.2.2 of Chapter 9)

- The plant designer will perform an analysis of anticipated containment airborne activity levels, estimated extent of personnel entries during normal plant operation, and radiation protection (ALARA) measures. This analysis will result in the establishment of the functional requirements for the containment cleanup and low-volume purge subsystems.

- The containment cleanup subsystem will be designed as a non-safety, non-seismic system powered from the permanent non-safety distribution system.

- The subsystem will be designed to provide air circulation and filtration to reduce the concentrations of airborne radioactivity for safe access during normal plant operation or after reactor shutdown. It will operate on an as-required basis for a predetermined number of hours.

Control Rod Drive Mechanism (CRDM) Ventilation Subsystem (Section 8.4.2.3 of Chapter 9)

- The subsystem will be designed as a non-safety system powered from the permanent non-safety power distribution system. It will be designed to withstand seismic loads in accordance with UBC, Zone 2A.

- The subsystem will be designed to establish air flow through the reactor head cooling shroud to maintain the CRDM temperature within the operating limits (determined by the reactor supplier) and to exhaust the air, after absorbing the heat, in the vertical direction.

- The subsystem will consist of operating and standby fans in accordance with the requirements of the reactor supplier.

- The subsystem will be controlled manually and monitored by status lights and motor trip alarms on the control room panel.

- The subsystem will be powered from the permanent non-safety distribution system on loss of offsite power.

Section 8.4.2.3 of Chapter 9 states that the CRDM ventilation subsystem should be designed to withstand earthquakes in accordance with UBC, Zone 2A. The provisions of UBC Zone 2A may be used for the CRDM ventilation subsystem design, provided the plant designer demonstrates that this subsystem will retain its structural integrity under the specified SSE. The staff will review individual application for FDA/DC to determine the acceptability of the plant designer's demonstration of the structural integrity of the CRDM ventilation subsystem under the specified SSE.

Reactor Cavity Cooling Subsystem (Section 8.4.2.4 of Chapter 9)

- The reactor cavity/ex-core instrumentation, reactor support and in-core instrumentation room ventilation subsystem will be designed as a non-safety, non-seismic system for operation during normal plant operation and loss of offsite power.

- Depending on the nuclear steam supply system design, this subsystem may be incorporated in the fan cooler subsystem.

- The subsystem will be designed to maintain temperatures within the following specified limits:

  - Reactor cavity average concrete temperature will be 150 °F with a maximum local area temperature of 200 °F.

  - Reactor support area temperature will be 135 °F with a maximum concrete temperature of 180 °F.

- In-core instrumentation room temperature will not exceed 120 °F.

- The subsystem will be provided with redundant divisions, each consisting of a supply fan, dampers, controls, and associated instrumentation.

- The subsystem will be powered from the permanent non-safety distribution system on loss of offsite power.

In the DSER for Chapter 9, the staff stated that in Section 8.4.2.4.2 of Chapter 9, the cooling coil supplied by chilled water was not included for the reactor cavity cooling subsystem. This omission was identified as an open issue. In its letter dated September 29, 1992, EPRI stated that Section 8.4.2.4.1 specifies the temperature range criteria in the area of reactor supports and reactor cavity concrete temperature. It is up to the plant designer to design a cooling system that meets the temperature criteria. The system may or may not have cooling coils. Therefore, the staff will evaluate the reactor cavity cooling subsystem design performance during its review of an individual application for FDA/DC to ensure that the temperature range criteria in Section 8.4.2.4.1 are met. Therefore, this open issue is closed.

Primary Containment Purge Subsystem (Section 8.4.2.5 of Chapter 9)

The primary containment purge subsystem will be designed as a non-safety system with the exception of containment penetrations that are designated as safety related. The ductwork inside the containment will be designed to seismic Category II/I criteria. The subsystem will be powered from the permanent non-safety distribution system and will consist of high-volume and low-volume purge subsystems as shown below:

(1) High-Volume Purge Subsystem

- The high-volume normal purge subsystem will be designed to supply 100-percent filtered, tempered (heated or cooled) outside air to maintain containment temperatures within the range given in Section 8.2.1.1.1 of Chapter 9 before and during personnel entry for cold-shutdown or refueling operations and to exhaust the same quantity of air through the exhaust filter system.

- The supply subsystem will consist of a prefilter bank, a high-efficiency filter bank, a heating coil, a cooling coil supplied by chilled water, and supply fans.

- The exhaust subsystem will consist of a prefilter bank, a high-efficiency particulate air (HEPA) filter, and a fan.

- The high-volume purge subsystem will be manually controlled from the control room. A local panel will also be provided with fan status indication lights and alarms actuated by high filter resistances.

- The containment isolation valves will be designed to withstand the loss-of-coolant accident (LOCA), integrated leak rate test pressures, and the high-pressure and humidity conditions generated by a main steam or feedwater pipe rupture, and to remain leaktight.

- A containment isolation signal will trip the supply and exhaust fans.

(2) Low-Volume Purge Subsystem

- The low-volume purge subsystem will be designed to supply 100-percent filtered air to the containment and in-core instrument room.
- The supply subsystem will consist of a heating coil, a prefilter bank, a high-efficiency filter bank, and a supply fan.
- The exhaust subsystem will consist of a prefilter bank, a HEPA filter, and an exhaust fan.
- The low-volume purge subsystem will be manually controlled from the control room. A local panel will also be provided with fan status indication lights and alarms actuated by high filter resistances.
- A containment isolation signal will trip the supply and exhaust fans.

Section 8.4.2.5 of Chapter 9 does not require charcoal filters for the containment purge subsystem. Should a LOCA occur during containment purging when the reactor is at power and before the purge subsystem in isolated, fission products may be released to the environment. However, in Section 8.2.1.1.26, EPRI states that the plant designer will perform analyses and evaluations to determine if charcoal filters are necessary to meet the requirements of 10 CFR Part 20 and 10 CFR Part 50, Appendix I, and to maintain in-plant ALARA levels in accordance with RG 8.8. The staff will evaluate the detailed system design operation and layout during its review of an individual application for FDA/DC to ensure that the guidelines of Branch Technical Position CSB 6-4 have been satisfied.

The staff concludes that the design requirements in Section 8.4.2 of Chapter 9 for the PWR containment cooling and ventilation system do not conflict with NRC regulatory guidelines and are, therefore, acceptable. The staff will evaluate the need for charcoal filters for this system on the basis of 10 CFR Part 50, Appendix I, during its review of an individual application for FDA/DC.

8.4.2 PWR Auxiliary Building Ventilation System

Section 8.4.3 of Chapter 9 states that the PWR auxiliary building ventilation system will consist of a normal ventilation subsystem and a filtered exhaust subsystem.

The key requirements for the auxiliary building normal ventilation subsystem are the following:

- The subsystem will be designed as a non-safety, non-seismic system supplied from the permanent non-safety distribution system and equipped with safety-related, seismically qualified isolation dampers.
- The subsystem will be designed to perform the following functions:

- maintain the general area temperature within the range given in Section 8.2.1.1.1 of Chapter 9

- maintain the ambient pressure slightly below atmospheric

- direct air flow from areas of lesser potential airborne contamination to areas of greater potential contamination

- provide ventilation, cooling, and heating to all equipment in the auxiliary building during normal plant operation

- provide filtered outside air to maintain the maximum permissible concentration levels in the cubicles within ALARA guidelines

- provide filtered outside air to general areas

- provide for a transit time of unfiltered exhaust air from the radiation monitors to the isolation dampers that is greater than the damper closing time plus the radiation monitor response time

- The supply subsystem will consist of redundant filter banks, heating coils, cooling coils supplied by chilled water, and supply fans.

- The exhaust subsystem will consist of exhaust fans to maintain the negative pressure in the building.

- The normal ventilation subsystem will be manually controlled from a local panel provided with temperature and pressure indicators and fan status lights.

- On receipt of an appropriate signal or a high-radiation signal at the unfiltered exhaust duct, the system design will ensure the following automatic actions: tripping of the normal ventilation subsystem, closing of the auxiliary building isolation dampers, and starting of the filtered exhaust subsystem.

The key requirements for the auxiliary building filtered exhaust subsystem are the following:

- The subsystem will be designed to withstand earthquakes in accordance with UBC, Zone 2A and be capable of accomplishing its intended function assuming a single active failure and loss of offsite power.

- The subsystem will be designed to perform the following safety-related functions during accident conditions:

  - exhaust the required quantity of air to maintain a slight negative pressure with respect to adjacent auxiliary building areas in the radioactive process areas and radioactive utility corridors

  - filter the exhausted air before it is discharged

- The subsystem will be designed to operate during normal plant operation to reduce undesirable levels of airborne contamination.

- The subsystem will be provided with redundant systems, each consisting of a bank of prefilters, a HEPA filter bank, an exhaust fan for maintaining the negative pressure, and associated dampers, ducts, instruments, and controls. The passive Requirements Document does not require the provision of charcoal filters for the auxiliary building filtered exhaust subsystem. However, in Section 8.2.1.1.26, EPRI states that the plant designer will perform analyses and evaluations to determine if charcoal filters are necessary to meet the requirements of 10 CFR Part 20 and 10 CFR Part 50, Appendix I, and to maintain in-plant ALARA levels in accordance with RG 8.8. The staff will review the detailed system design operation and layout during its review of an individual application for FDA/DC to ensure that the guidelines of Branch Technical Position CSB 6-4 have been satisfied.

- The subsystem control will be designed for automatic initiation on receipt of an appropriate signal, and manual start from the control room.

- A local control panel will be provided with switches, fan status lights, instrumentation, and alarms.

The passive Requirements Document indicates that, following a LOCA, only a small amount of air will be exhausted from the auxiliary building to achieve a slight negative pressure within the auxiliary building. The amount of makeup air required will be drawn in through some predetermined openings. Therefore, EPRI states that the supply subsystem will not be needed and will not be designed as safety related. Because of the flexibility in system design, the staff will evaluate the detailed system design and operation and facility layout during its review of an individual application for FDA/DC to determine the need for a safety-grade supply system.

The staff concludes that the design requirements in Section 8.4.3 of Chapter 9 of the passive Requirements Document for the PWR auxiliary building ventilation system do not conflict with the guidance in SRP Sections 9.4.3 and 9.4.5 and are, therefore, acceptable.

8.5 Conclusion

In the DSER for Chapter 9, the staff stated that many requirements in Sections 8.2 and 8.3 of this chapter did not explicitly require the HVAC systems to meet the SRP, the GDC, regulatory guidance, or Commission regulations. The staff provided multiple examples in Sections 8.2.5 and 8.3.2 of this chapter of the DSER. In the DSER, the staff concluded that EPRI should provide sufficient justification for these requirements, demonstrate how they meet the appropriate regulations or regulatory guidance, or modify them to require compliance with the appropriate regulations or regulatory guidance. This was identified as an open issue. In its letter dated September 29, 1992, EPRI stated that the passive plant HVAC systems are non-safety grade and are not required to perform any safety-related functions. Therefore, EPRI stated that this open issue should be resolved as part of the policy issue associated with RTNSS. The regulatory treatment of non-safety HVAC systems will be determined in accordance with the resolution of Policy Issue III.A, which was identified in Annex A of Appendix B to Chapter 1 of the DSER. In SECY-94-084, provided as Annex B of this Appendix, the staff describes a process for resolving Policy Issue III.A. In this process, the ALWR designer will identify all

risk-significant non-safety SSC functions and their corresponding R/A missions. Furthermore, the designer will propose regulatory treatment appropriate to provide confidence that the identified R/A mission will be achieved. In its SRM dated June 30, 1994, the Commission approved the staff's recommendation on RTNSS. The staff will evaluate the results of this process during its review of an individual application for FDA/DC to determine if enhanced regulatory oversight is required for any of these systems. Therefore, this open issue is closed.

In several instances, EPRI states that all of the control complex HVAC systems and components are classified as non-safety items and will be designed for seismic loads in accordance with UBC, Zone 2A. This design requirement is specified in Sections 8.2.2, 8.2.4, 8.3.4, 8.4.2.3, 8.4.3.2, and 8.4.3 of Chapter 9 of the passive Requirements Document. In the DSER for Chapter 9, the staff stated that, because the use of this code might not be adequate to provide the needed reliability of these systems, EPRI should justify its use and identified this as an open issue. In its letter dated September 29, 1992, EPRI stated that according to the classification requirements in Section 4.3 of Chapter 1 of the passive Requirements Document, the non-safety items should include both seismic Category II and non-seismic structures, systems, and components, and the seismic Category II items should be designed to ensure that they will not fail during and after an SSE or the structural failure of these items will not degrade the functioning of a seismic Category I item and will not result in incapacitating injury to occupants of the control room. As for the design of the control complex HVAC systems and components, on the basis of the resolution of the issue in Chapter 6 of this report concerning turbine-generator building SSE loading condition, the use of UBC Zone 2A is acceptable, provided the plant designer demonstrates that the control complex HVAC systems and components will retain their structural integrity and that there will be no interaction with seismic Category I items under the specified SSE. The staff will review an individual application for FDA/DC to ensure that this criterion is met. Therefore, this open issue is closed.

In addition, as noted in the DSER, in Sections 8.2.2, 8.2.3, 8.3.4, and 8.4.4 of Chapter 9, EPRI did not specify power sources or adequately describe how the system or component will perform following a loss of offsite power. This was identified as an open issue. In its letter dated September 29, 1992, EPRI stated that all HVAC systems are categorized as non-safety and are used to maintain air quality environment. Their "permanent non-safety loads" are fed from either a permanent non-safety bus or by one of two standby generators in the event of a loss of offsite power as defined in Chapter 11, Section 2.3.2, and as shown in Figures 11.2-1 and 11.2-2. The staff will review the results of the process outlined in SECY-94-084 during its review of an individual application for FDA/DC to determine if enhanced regulatory oversight is required for any of these systems. Therefore, this open issue is closed.

The requirements in Section 8 of Chapter 9 of the passive Requirements Document, by themselves, do not provide sufficient information to make a determination that the plant-specific design and arrangement will be adequate. Therefore, applicants referencing the passive Requirements Document will be required to demonstrate compliance with the additional guidance in the SRP, or provide justification for alternative means of implementing the associated regulatory requirements.

9 LABORATORIES

Section 9 of Chapter 9 of the passive Requirements Document given the requirements for the laboratories (including counting rooms, cold chemistry facilities, and calibration facilities) of an ALWR plant. The primary functions of these laboratories are to provide (1) plant support services for routine health physics analysis, (2) normal and postaccident cold chemical analysis of required plant chemistry samples, (3) routine and postaccident counting of plant radioactivity samples, (4) grab sample analysis, and (5) a facility to store and secure radioactive calibration and check sources and instruments.

Section 9.2 of Chapter 9 states that the laboratories will be sufficiently equipped to permit the onsite performance of all critical onsite sampling specified in RG 1.97. The passive Requirements Document includes the specifications for the plant sampling and laboratory facilities where normal and postaccident cold chemical and radioactive analyses are to be performed. It requires that the plant designers ensure that suitable laboratory services, service connections, support equipment, and instrumentation are provided at the plant site. Moreover, Section 9.6 of Chapter 9 specifies that a computer system should be provided for evaluating the collected data. The document also requires that a maintenance program be established for all the sampling and analytical facilities. EPRI states that the plant designer will provide a detailed description of the equipment and instrumentation to be used in these laboratories. To maintain radiation doses to laboratory personnel as low as is reasonably achievable and to ensure reliable instrument radiation readings, radiation counting rooms, instrument calibration areas, and checkout areas will be located in low-radiation areas of the plant. Section 9.4 of Chapter 9 states that laboratory check and calibration sources will be secured in shielded storage areas to minimize radiation doses to laboratory personnel and to ensure that low background radiation levels are maintained in counting room areas.

In Revision 2 of Appendix B to Chapter 1 of the passive Requirements Document, EPRI commits to comply with 10 CFR 50.34(f)(2)(xxvii), which addresses Item III.D.3.3 of NUREG-0737 on postaccident iodine sampling. Item III.D.3.3 of NUREG-0737 states that licensees must provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas in the facility where plant personnel may be present during an accident. However, the listing of specific equipment to be used to measure plant airborne iodine concentrations is outside the scope of the passive Requirements Document and will be provided by the plant designer.

The staff concludes that the functions and design features of the laboratories comply with the guidance in RG 8.8 and SRP Section 12.5, "Operational Radiation Protection Program," and the Commission guidelines regarding the chemical engineering and radiation protection aspects related to the design and operation of the decontamination and normal and postaccident sampling systems and are, therefore, acceptable. However, by themselves, they do not provide sufficient information to make a determination that the plant-specific design and arrangement will be adequate. Therefore, applicants referencing the passive Requirements Document must demonstrate compliance with the additional guidance in the SRP, or provide justification for alternative means of implementing the associated regulatory requirements.

10 CONCLUSION

The staff concludes that the EPRI requirements established in Chapter 9 of the passive Requirements Document for the design and arrangement of site support systems do not conflict with current regulatory guidelines and are acceptable. However, by themselves, they do not provide sufficient information for the NRC staff to determine if the plant-specific design and arrangement of site support systems will be adequate. Applicants referencing the ALWR passive Requirements Document will be required to demonstrate compliance with the additional guidance in the SRP, or provide justification for alternative means of implementing the associated regulatory requirements.

Therefore, the staff concludes that Chapter 9 specifies requirements that, if properly translated into a design in accordance with the NRC regulations in force at the time the design is submitted, should result in a nuclear power plant whose site support systems will perform as designed and have all the attributes required by the regulations to ensure that there is no undue risk to the health and safety of the public or to the environment.

APPENDIX A
DEFINITIONS AND ACRONYMS

Appendix A of Chapter 9 of the passive Requirements Document contains definitions of acronyms. The staff has provided a consolidated list of acronyms in Volume 1 of this report.

CHAPTER 10, "MAN-MACHINE INTERFACE SYSTEMS"

1 INTRODUCTION

This chapter of the SER documents the NRC staff's review of Chapter 10, "Man-Machine Interface Systems," of the passive Requirements Document through Revision 5. Revision 5 was submitted to the NRC by letter dated October 1, 1993. Chapter 10 was prepared, under the project direction of EPRI and the Advanced Light-Water Reactor (ALWR) Utility Steering Committee, by Duke Power Company; General Electric Company; MPR Associates, Inc.; S. Levy Incorporated; Science Applications International Corporation; Westinghouse Electric Corporation; and EPRI.

1.1 Review Criteria

Section 1 of Volume 1 of this report and Section 1.4 of this chapter describe the approach and review criteria used by the staff during its review of Chapter 10 of the passive Requirements Document.

1.2 Scope and Structure of Chapter 10

Chapter 10 of the passive Requirements Document defines the ALWR Utility Steering Committee's requirements for the design of the man-machine interface systems (M-MIS) for passive ALWRs.

The staff notes that the majority of the requirements in Chapter 10 of the Requirements Document for passive plants are identical to the requirements in Chapter 10 of the Requirements Document for evolutionary plants. The major exceptions are the requirements in Sections 7 and 8 due to specific system functional and performance requirements in the passive plant and the differences in dc and low voltage ac power supply requirements between evolutionary and passive plant designs.

The requirements of Chapter 10 involve, but do not repeat, the requirements from other chapters. The potential problem that arises from this document structure is discussed in Section 1.4.2, below.

Section 1, "Introduction," summarizes the scope of the requirements for the M-MIS and the interfaces of the M-MIS with the system addressed in other chapters.

Section 2, "Objectives and Policy," contains the specific M-MIS design requirements that EPRI has provided to implement the general policy statements presented in Volume I of the Requirements Document. This section provides overall direction and guidance for the M-MIS design development and implementation and addresses specific issues that have arisen out of operating experience with present M-MIS.

Section 3, "Key Requirements," provides top-level design requirements for the M-MIS that enforce the objectives and policy statements in Section 2. These requirements cover the M-MIS design organization, the responsibilities of the M-MIS designer, the process for design development and implementation, and the verification and validation (V&V) of the M-MIS design.

Section 4, "Control Station Requirements," contains requirements for control facilities staffing, control facilities, and information displays, plant controls, and alarms.

Section 5, "Data Gathering, Transmission and Processing Requirements," contains requirements for the acquisition, processing, and distribution of the large amounts of data used for plant protection, control and monitoring, that is, the advanced distributed data system that is at the heart of the M-MIS. It provides both hardware and software requirements for the necessary equipment such as computers, multiplexers, fiber optic and hardwired data transmission, and self diagnostics, as well as design requirements pertaining to overall functions, numbers of channels, data rate, failure modes, etc.

Section 6, "Common Software, Hardware and Control Requirements," contains the common software, hardware and control requirements for design, implementation, and installation of the M-MIS.

Section 7, "Overall Plant, Reactor and Reactor Coolant Systems," contains requirements for specific instrumentation and control (I&C) systems that are based on the plant protection, control, and monitoring functions performed by the systems.

Section 8, "Reactor Protection and Safety Systems," contains requirements for specific I&C systems that are based on the plant protection, control, and monitoring functions performed by the systems.

Section 9, "Power Generation and Main Turbine-Generator Systems," contains requirements for specific I&C systems that are based on the plant protection, control, and monitoring functions performed by the systems.

Section 10, "Auxiliary and Plant Support Service Systems," contains requirements for specific I&C systems that are based on the plant protection, control, and monitoring functions performed by the systems.

1.3 Policy Issues

In staff requirements memoranda (SRM), the Commission instructed the staff to provide an analysis detailing where the staff proposes departure from current regulations or where the staff is substantially supplementing or revising interpretive guidance applied to currently licensed light-water reactors (LWRs). During its review of Chapter 10 of the passive Requirements Document, the staff did not identify any policy questions for the technical areas discussed in this chapter, other than those already identified in the Commission papers listed in Appendix B to Chapter 1 of this report.

Two recent policy issues pertain to Chapter 10, both of which are included in the SECY-93-087, "Policy, Technical and Licensing Issues Pertaining to Evolutionary and Passive Light-Water Reactor Designs," issued on April 2, 1993.

One policy issue, defense against common-mode failures in digital I&C systems, is addressed in Section II.Q of SECY-93-087 and is provided in Annex A of Appendix B to Chapter 1 of this report. The staff proposed that applicant for FDA/DC be require to perform an assessment of the defense-in-depth and diversity of the proposed I&C system to demonstrate that vulnerabilities to common-mode failures have been adequately addressed, including requirements for performing the assessment and the need for a diverse means to counter the identified common-mode failures. In its staff requirements memorandum (SRM) dated July 21, 1993, the Commission approved the staff's proposed requirement for an assessment of the defense-in-depth and diversity of proposed I&C systems at ALWRs. However, the Commission directed that the assessment may be performed using best-estimate methods. The Commission also clarified that if a postulated common-mode failure that could disable a safety function is identified, then the diverse means required to perform either the same function or a different function may be performed by a non-safety system of the system is of sufficient quality. In SECY-93-087, the staff also proposed the requirement for a set of safety-grade displays and controls, independent and diverse from the computer systems identified in the above assessment, for manual system level actuation of critical safety functions and monitoring of parameters that support the safety functions. In the July 21, 1993, SRM, the Commission approved the requirement for displays and controls in the main control room for manual, system-level actuation of critical safety functions and monitoring of parameters that support the safety functions. However, the Commission provided additional guidance that such I&C equipment need not be safety-grade.

The second policy issue, the regulatory treatment of non-safety systems in passive plant designs, is addressed in SECY-94-084 (contained in Annex B of Appendix B to Chapter 1 of this report). The applicable part of this policy issue to Chapter 10 is the recommendation for a graded safety classification system for I&C systems that perform functions that are important to safety but that are not part of the safety system. The subject of a graded safety classification system for I&C systems is discussed in SECY-91-292, "Digital Computer for Advanced Light Water Reactors," dated September 16, 1991. A partial response to this recommendation for a graded safety classification system for I&C systems has been incorporated in the passive Requirements Document. However, this addition to the passive Requirements Document was in response to a draft safety evaluation report (DSER) open issue on software requirements, not in response to the policy issue. Because the staff has not yet published its final position on the matter of graded requirements, based on the importance to safety of the function(s) performed by I&C systems and equipment, it has not been addressed by EPRI in the passive Requirements Document.

In the February 15, 1991, SRM pertaining to SECY-90-377, "Requirements for Design Certification Under 10 CFR Part 52," the Commission directed the staff to review the Requirements Document to ensure that it was sufficient to allow the staff to evaluate the severe-accident issues and the incorporation of experience from operating events in current designs.

The staff concludes that the information in the I&C scope of review does not specifically address any severe-accident issues. The M-MIS, as designated by EPRI, provide support functions to the primary systems. Information in the M-MIS area that will be required for evaluating the severe-accident issues, such as reliability of automatic or manual actions, is not provided.

1.4 Outstanding Issues

1.4.1 Outstanding Issues From Staff's DSER for Chapter 10 of Passive Requirements Document

The DSER for Chapter 10 of the passive Requirements Document contained the following outstanding issues:

Open Issues

P.10.0-1 independence of the software verification and validation review teams (3.1.2, 3.1.4, and 6.1.5)

P.10.0-2 use of commercial compilers for software used in safety systems (3.1.2)

P.10.0-3 dedication of commercial-grade software (3.1.2)

P.10.0-4 clarification of requirements for analysis and validation testing of M-MIS (3.1.3)

P.10.0-5 use of rapid prototyping in the development and validation of functional specifications (3.1.3)

P.10.0-6 operator aids (3.4.5)

P.10.0-7 establishment and use of reliability and availability estimates (3.5)

P.10.0-8 definition of practical (3.5.3)

P.10.0-9 environmental conditions for equipment design including compatibility with tasks (3.5.3)

P.10.0-10 component reliability of M-MIS (3.5.4)

P.10.0-11 overall reliability of M-MIS (3.5.4)

P.10.0-12 sneak circuit analysis (SCA) (3.5.4)

P.10.0-13 software maintenance (3.5.4)

P.10.0-14 minimum tests for continuous on-line testing (3.6.1)

P.10.0-15 vulnerability of power supplies for alarm systems (4.3.1)

P.10.0-16 guidance on criteria to establish priorities (4.3.4)

P.10.0-17 guidance on the maximum number of alarms (4.3.4)

P.10.0-18 alarm sequence recording (4.3.4)

P.10.0-19 valve position indication (4.4)

P.10.0-20 guidance on frequency allocation plan (4.6)

P.10.0-21 guidance on interference between communication systems and M-MIS equipment (4.6)

P.10.0-22 environmental conditions for minimally used local control stations (4.9.2)

P.10.0-23 guidance on inadvertent actuation of controls at local control stations (4.9.2)

P.10.0-24 special physical security measures for the data transmission cable (5.2.1)

P.10.0-25 guidance on data system characteristics (5.2.2)

P.10.0-26 propagation of common-mode failures through the data system (5.2.2)

P.10.0-27 expansion capability of multiplexers (5.2.3)

P.10.0-28 reliability of multiplexing system (5.4)

P.10.0-29 software design aids and tools (6.1.1 and 6.1.5)

P.10.0-30 quality assurance requirements for safety-related software (6.1.2)

P.10.0-31 configuration-management requirement for software (6.1.2)

P.10.0-32 software integrity (6.1.2)

P.10.0-33 guidance on software user documentation (6.1.2)

P.10.0-34 acceptance testing of commercially available software (6.1.2)

P.10.0-35 notification of software errors or modifications of commercially delivered software products (6.1.2)

P.10.0-36 long-term configuration control of software (6.1.2)

P.10.0-37 clarification of top-down structured design approach (6.1.3)

P.10.0-38 guidance on convolution of software structure (6.1.3)

P.10.0-39 behavior of commercial software when assumptions are violated (6.1.3)

P.10.0-40 guidance on memory protection (6.1.3)

P.10.0-41 use of information by redundant safety channels (6.1.3)

P.10.0-42 guidance on tools to improve reliability and quality of software (6.1.5)

P.10.0-43 definition of reasonable testing and degree of confidence to be considered (6.1.5)

P.10.0-44 specification of the level of diversity in safety systems (6.1.6, 6.2.3, and 6.2.5)

P.10.0-45 guidance on performance of reliability evaluation (6.1.6)

P.10.0-46 reference to Institute of Electrical and Electronics Engineers (IEEE) 1050-1989, "IEEE Guide for Instrumentation and Control Grounding in Generating Stations" (6.2.2)

P.10.0-47 compatibility between M-MIS equipment and its external power supply systems (6.2.2)

P.10.0-48 alarmed, self-diagnostic feature on clock update (6.2.3)

P.10.0-49 guidance on position of sensor isolation valves (6.2.5)

P.10.0-50 capacitance-type pressure sensors (6.2.5)

P.10.0-51 minimal acceptance review criteria for isolation device (6.2.6)

P.10.0-52 voltage design of battery and dc system (6.2.8)

P.10.0-53 standards for surge suppression (6.2.8)

P.10.0-54 electromagnetic interference/radiofrequency interference (EMI/RFI) considerations for wiring shields (6.2.9)

P.10.0-55 specific grounding standards (6.2.9)

P.10.0-56 use of qualified isolators for wiring shields (6.2.9)

P.10.0-57 requirements for signal reconstruction (6.3.3)

P.10.0-58 use of interrupts (6.3.3)

P.10.0-59 redundancy of safety systems (6.3.4)

P.10.0-60 selection of automatic or manual control (7.2)

P.10.0-61 operation of plant by load dispatcher (7.3)

P.10.0-62 alternate means of inventory monitoring (7.13)

P.10.0-63 clarification of actuation logic self testing (8.2.3)

P.10.0-64 initiation of standby liquid control system (8.7)

P.10.0-65 inclusion of four elements in human factors organizational structure (3.1.2 of Appendix B)

P.10.0-66 scheduling of human factors studies before start of control room design process (3.1.3 of Appendix B)

P.10.0-67 guidance to the M-MIS designer to overcome past problems (3.1.4 of Appendix B)

P.10.0-68 guidance to improve interfaces between the operator and the plant (3.1.4 of Appendix B)

P.10.0-69 identification of human factors criteria, guidance, etc., which were used as the supporting bases for M-MIS requirements (3.2.1 of Appendix B)

P.10.0-70 traceability of human factors requirements to original source (3.2.2 of Appendix B)

P.10.0-71 method for establishing effective human factors requirements (3.2.3 of Appendix B)

P.10.0-72 guidance on systems analysis (3.2.4 of Appendix B)

P.10.0-73 organization of plant information (3.2.4 of Appendix B)

P.10.0-74 configuration of operator's workstation (3.2.5 of Appendix B)

P.10.0-75 human factors guidelines for new technology (3.3 of Appendix B)

P.10.0-76 illumination levels (3.3 of Appendix B)

P.10.0-77 maintenance procedures (3.4 of Appendix B)

P.10.0-78 selection and qualification of plant personnel (3.5 of Appendix B)

P.10.0-79 training requirements for top-level personnel (3.5 of Appendix B)

P.10.0-80 human factors verification and validation test plan (3.7.1 of Appendix B)

P.10.0-81 documentation of human factors test activities (3.7.2 of Appendix B)

P.10.0-82 team performance (3.7.4 of Appendix B)

Confirmatory Issues

P.10.C-1 intention of use (1.3)

P.10.C-2 impact of support software on installed software (6.1.2)

P.10.C-3 software common-mode failures (6.1.6 and 6.2.5)

P.10.C-4 boiling water reactor (BWR) depressurization system design change (8.6)

P.10.C-5 staffing of M-MIS that controls security functions (10.2.3)

The final disposition of each of these issues is discussed in detail in the appropriate section of this chapter, as indicated by the parenthetical notation following each issue. All outstanding issues identified in the DSER for Chapter 10 of the passive Requirements Document have been resolved.

1.4.2 Resolution of Concerns Identified in Part I of Staff's DSER for Chapter 10

In Part I of the DSER for Chapter 10 of the passive Requirements Document, the staff identified several concerns related to the two major differences between the passive plant M-MIS and the M-MIS that are currently in use at operating plants. One difference is the application of modern digital technology to implement the protection, control, and monitoring functions of the ALWR for both the evolutionary and passive plant designs. The other difference is the reliance on what are categorized as non-safety systems to perform functions that are intended to ensure safe shutdown and avoid the need to initiate passive safety systems.

At present, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), does not contain detailed acceptance criteria for designs of the I&C systems and equipment that are different from those at operating plants. In its review of Chapter 10 of the passive Requirements Document, the staff augmented the standard review plan (SRP) with guidance based on Enclosure 2 of SECY-91-292.

Several concerns pertaining to the M-MIS were identified in Part I of the DSER for Chapter 10, based on Enclosure 2 of SECY-91-292. These concerns, separate from the open issues identified in the remainder of the DSER for Chapter 10, were grouped into five categories:

- system integrity
- software integrity
- hardware integrity
- data system integrity
- backup safety system

System Integrity

Three specific concerns regarding system integrity were identified in Part I of the DSER for Chapter 10.

(1) In the DSER, the staff concluded that the passive Requirements Document should include consideration of rapid prototyping in the M-MIS design process to help ensure the completeness of the system specification and the feasibility of the design concept. In Revision 4, Section 3.1.3.5.1 was added requiring rapid prototyping as part of the M-MIS design process. This provided guidance that the staff concluded was necessary and, therefore, resolved this concern.

(2) In the DSER, the staff concluded that a graded classification system was needed to identify systems that perform functions important to safety. This concern is included in the policy issue associated with the regulatory treatment of non-safety systems. As described in SECY-94-084, which is provided in Annex B of Appendix B to Chapter 1 of this report, the staff will evaluate the manner in which passive plant vendors follow the process outlined in EPRI's letter dated May 26, 1993, during its review of an individual application for final design approval/design certification (FDA/DC). A key element of this process is to ensure that passive plant vendors use probabilistic risk assessment methods to examine the reliability of the safety and non-safety systems and ensure that important risk insights are considered in the design. In addition, in Revision 4, EPRI added requirements to Section 6.1.8 to include graded requirements for software. This addition provided the guidance for the software design process that the staff had requested; therefore, this concern is resolved.

(3) In the DSER, the staff concluded that EPRI had not included adequate defense-in-depth requirements for the M-MIS. This concern is covered by the policy issue dealing with defense against common-mode failures in digital I&C systems and will be resolved with the individual passive plant vendors during the staff's review of an individual application for FDA/DC.

Software Integrity

Three specific concerns regarding software integrity were identified in Part I of the DSER for Chapter 10.

(1) In the DSER, the staff concluded that the requirements should call for the use of modern software development processes. In Revision 4, Section 6.1.2.22 was added, requiring the M-MIS designer to use modern software development techniques and practices and presenting examples of modern software development methods that should be considered. The staff concludes that this provides the needed guidance for modern software development processes.

(2) In the DSER, the staff concluded that the requirements in Section 6.1 should be graded with respect to importance to safety. In Revision 4, EPRI added Section 6.1.8, "Applicability of Software Requirements," and revised Section 6.1.2.1 to clarify the software quality assurance program (SQAP) for non-safety software. The staff concludes that this is a significant step forward in dealing with this issue for software. Therefore, this concern is resolved for the passive Requirements Document.

A more comprehensive categorization of the different levels of importance to safety is needed to provide the basis for system and equipment design requirements. This subject is part of the policy issues on both the regulatory treatment of non-safety systems and the defense against common-mode failures in digital I&C systems and will be resolved with passive plant vendors during the staff's review of an individual application for FDA/DC.

(3) In the DSER, the staff concluded that requirements for building confidence in the software after completion of the software development process for functions ranked in the higher levels of importance to safety should be added. The resolution proposed by EPRI is based on activities performed as part of the development process and not activities performed after the completion of the software development process. The purpose of confidence building is to provide additional evidence that the final software product performs as intended, making use of test environments different from those used in the development process. One could consider obtaining this evidence as part of the commissioning program. This would provide a test of the M-MIS in their various configurations for the different phases of the startup test programs. The staff will resolve this matter with the vendors during its review of individual applications for FDA/DC.

In addition to these three concerns, two requirements in the passive Requirements Document that could result in an increase in software complexity were identified.

In Section 3.4.5 of the DSER for Chapter 10, the staff expressed concern that the requirement for operator aids to be considered an integral part of the M-MIS design, while correct from a human factors view, could lead to an increase in safety system software complexity without some limitation on this requirement. In Revision 5, EPRI added Section 6.1.3.7.1 requiring that for safety-related software, complexity be minimized to the extent practical.

In Section 3.6.1 of the DSER for Chapter 10, the staff expressed a concern that the continuous on-line testing requirement in Section 3.6.1 of Chapter 10 could also lead to a potential increase in safety system software complexity. In Revision 5, EPRI added Section 6.1.3.7.1 requiring that for safety-related software, complexity be minimized to the extent practical.

The staff concludes that these changes are acceptable and resolve the staff's concern about increases in software complexity.

Hardware Integrity

One specific concern regarding hardware integrity was identified in Part I of the DSER for Chapter 10.

In the DSER, the staff concluded that EPRI should clarify the requirements for loss of environmental control for 72 hours versus loss of environmental control for individual cabinets (i.e., without an operable heating, ventilating, and air conditioning system (HVAC)).

In Revision 4, EPRI modified Section 3.5.3.3 to provide additional clarification by requiring that M-MIS equipment be designed to survive and meet all of the reliability and availability requirements in the environment in which it is located for both normal and off-normal conditions. EPRI also changed Section 5.3.3.2 by requiring that all M-MIS equipment, including the quality level of the components that is not required to function following a loss of both onsite and offsite ac power be designed to cope with a loss of environmental control for 1 hour. The staff concludes that these clarifications resolve the above concern.

Data System Integrity

Two specific concerns regarding data systems integrity were identified in Part I of the DSER for Chapter 10.

(1) In the DSER, the staff concluded that special measures were necessary to guard against accidents and tampering of data transmission cables. For example, use of conduit for data transmission cables would mitigate the disruption to the M-MIS should data transmission cables be severed or otherwise rendered inoperable.

In its letter dated September 29, 1992, EPRI restated the existing requirements for segmentation of major functions and the requirements for redundancy and separation of safety and non-safety cables and proposed an additional requirement, Section 6.2.11.1.7, that data transmission cables for the M-MIS meet the separation criteria in IEEE 384 (1981), Section 6. There is still a need to be more specific regarding the requirements for the alternative design that addresses the situation described in the rationale for Section 3.5.3.1, which states,

> It is recognized that there may be situations in which strict segmentation may not be practical. In those few instances when it is not practical, alternate design approaches are allowed. Depending solely on redundancy is not allowed since: (1) a failure in one of the redundant paths could cause a heavy workload and possible confusion for the operator by indicating there is a problem in more than one major function, and (2) unforeseen or unexpected failures (e.g., maintenance errors) which affect more than one redundant path could cause large-scale upsets threatening more than one major process function.

The staff will resolve this concern with the passive plant vendors during its review of individual applications for FDA/DC.

(2) In the DSER, the staff concluded although EPRI addresses the requirements of redundancy and diversity in the data system through reference to Sections 3 and 6 of Chapter 10, these sections do not adequately cover the need for requirements that specifically guard against the propagation of common-mode failures through the data system. In Revision 5, EPRI added data systems to the list of M-MIS features that require defense-in-depth assessment of common-mode failures in Section 3.5.3.2. This provided the needed guidance; therefore, this concern is resolved.

Backup Safety System

No specific open issues were identified in Part I of the DSER concerning the backup safety system; however, concerns on this subject were the basis for the policy issue on defense against common-mode failures in digital I&C systems.

Section 2.2.10 contains a general requirement that "certain controls important to safety shall be provided in a separate and diverse manner (e.g., conventional, hardwired controls). These controls will provide a diverse backup

Manual Actuation of Safety Functions, requires that these be manual system-level actuation controls in the MCR, and states that these manual controls meet specific requirements including:

- These controls shall be physically separated from other controls in the MCR and shall meet applicable criteria for physical separation from each other.

- These controls shall be conventionally hardwired to as low a level in the M-MIS architecture as is practical and still achieve system level actuation.

The requirement for the manual actuation does not, however, specify that the control logic associated with the system level manual actuation be independent and diverse from the safety computer system. As discussed previously, the staff's position, as modified by the July 21, 1993, SRM, is that independent and diverse control equipment is required and that it need not be safety-grade, provided the control equipment is of sufficient quality to perform the necessary function under the associated event conditions. Therefore, the staff will review individual applications for FDA/DC to ensure that the design includes independent and diverse control equipment of sufficient quality.

Section 2.2.10 also contains a general requirement that:

> A series of non-redundant, integrated upright displays and alarms shall be placed on or near MCR walls to provide a mimic with plant status information for essential equipment operated from the MCR and key parameter values such as water levels, temperature, pressure, power levels and the presence of flow. The alarms will provide a set of top level, spatially dedicated alarms visible throughout the MCR. The large, upright displays and alarms will provide a spatially dedicated, continuously viewable, integrated mimic presentation of the plant status in a direct manner to a level of detail beyond that of summary information to enable the operators to confidently assess the status of essential equipment operated from the MCR.

This requirement does not specify that these displays shall be independent and diverse from the safety computer system. The staff's position is that if these displays are to be used for manual system-level actuation of critical safety functions and monitoring of parameters that support the safety functions they shall be independent and diverse from the safety computer system. Therefore, the staff will review individual applications for FDA/DC to ensure that the design includes independent and diverse display equipment.

Safety Credit for Non-Safety Systems

In Section 2.3 of Chapter 3, EPRI lists non-safety auxiliary systems that provide additional "defense-in-depth" capability for reactor coolant makeup and decay heat removal. These systems are in accord with the policy statement in Chapter 3, Section 1.5.2, "Defense-in-Depth Role of Non-Safety Auxiliary Systems."

The functional and performance requirements for these systems are given in various sections of different chapters of the passive Requirements Document. Section 7 of Chapter 10 acknowledges the existence of these systems, but provides little information for their design and qualification. In the DSER for Chapter 10, the staff identified a concern regarding the completeness and accuracy of the system specification, along with the need for a classification system that recognized different levels of importance to safety based on the functions performed by the systems. The staff concluded that graded requirements consistent with the level of importance to safety should be provided. The policy issue for defense against common-mode failures states that "the diverse or different function may be performed by a non-safety system if the system is of sufficient quality to perform the necessary function under the associated event conditions." In other words, the system is designed to an agreed set of standards and qualified for the specified service conditions. This matter will be resolved with the individual passive plant vendors during the staff's review of individual applications for FDA/DC.

Quantitative Availability and Reliability Criteria

Section 3.5 of Chapter 10 states that "the M-MIS designer shall establish quantitative reliability and availability criteria for the component parts and subsystems for the M-MIS which are consistent with the overall ALWR requirements." Unfortunately there are no agreed-on means to provide a reliability prediction for subsystems and even components that are a combination of hardware and software, where the software is the instruction procedure that directs the function of the hardware, except in almost trivial cases. This results from the lack of a means to predict limitations on the reliability of these subsystems and components caused by design errors, particularly for the software. One can, however, use experience data to predict the reliability of subsystems based on hardware random failures, for the case of hardware that is based on mature technology, provided one does not claim a random failure rate that exceeds the limitations placed by design errors related to the specific application.

The result of this situation is that one can use conventional reliability prediction and analysis techniques where the hardware random failure rate is the dominant factor, such as mean failure rates of components for the purpose of maintenance-related activity analyses; however, one needs to seek an alternative approach to determine consistency with very low risk system goals, such as core-melt probabilities.

Consequently, if the reliability and maintainability analyses in Section 3.5.4 are based on random equipment failures and limited to human errors caused by mistakes rather than human errors made as a result of wrong decisions, the regulatory concerns in this area are resolved. However, a different approach is needed to achieve the purpose stated in Section 3.5. This matter will be resolved with each individual passive plant vendor during the staff's review of the individual application for FDA/DC.

Organization of EPRI Requirements Document

The EPRI Requirements Document is organized according to an inheritance hierarchical approach, where once a requirement is stated it is inherited by any following applicable sections. However, the linkages between the parent

section and the subsequent applicable sections or vice-versa are not clearly expressed. This lack of a linkage map increases the probability that the specifications for a particular M-MIS will not be complete, resulting in changes in the specifications in the latter stages of design, if at all, with the resulting increase in the probability of design errors. Omissions and inaccuracies in the system specification are given as a major source of software errors; however, this situation can also result in hardware problems, such as the specification of the environmental service conditions for equipment.

This inheritance hierarchical structure is analogous to the organization used in object-oriented technologies, except that for object-oriented technologies a clear linkage of the hierarchical structure is given; this serves as a map between the different classes and objects so that linkages are provided for requirements that are expressed in one object but are also applicable to other objects. Several electronic textual systems exist that can help alleviate this problem, such as the PC-based hypertext and object-oriented database programs.

1.4.3 Conclusion

All outstanding issues pertaining to Chapter 10 of the passive Requirements Document have been resolved.

1.5 Vendor- or Utility-Specific Items

The DSER for Chapter 10 contained the following vendor- or utility-specific items. The final disposition of these items is discussed in the appropriate sections of this chapter, as indicated by the parenthetical notation following each item.

P.10.V-1 acceptable interpretations of requirements (1.3)

P.10.V-2 software protection (2.3)

P.10.V-3 level of automation (2.3)

P.10.V-4 defense-in-depth and diversity analysis (2.3 and 4.5)

P.10.V-5 review of equipment used for displays to the operator (2.3)

P.10.V-6 methods to ensure operator alertness (2.3)

P.10.V-7 additional criteria for developing technology (2.3)

P.10.V-8 independence of verification and validation review teams (3.1.2)

P.10.V-9 use of commercial-grade equipment (3.1.2)

P.10.V-10 complexity of M-MIS (3.1.3)

P.10.V-11 use of unproven technology (3.2.2)

P.10.V-12 quantitative reliability criteria (3.5)

P.10.V-13 selection of equipment failure modes (3.5.1 and 6.2.7)

P.10.V-14 maintenance frequency (3.5.2)

P.10.V-15 reliability analysis (3.5.4)

P.10.V-16 automatic reconfiguration after failure detection (3.6.4)

P.10.V-17 surveillance period of automatic testing features (3.6.8)

P.10.V-18 automatic bypass initiation (3.6.10, 3.6.13, and 3.6.14)

P.10.V-19 module software concerns (3.7.4)

P.10.V-20 bypass and test lockouts during on-line repairs (3.7.6)

P.10.V-21 MCR staffing (4.2)

P.10.V-22 alarm suppression techniques (4.3.3)

P.10.V-23 use of "dial-up" telephone-type portable radios for security purposes (4.6)

P.10.V-24 unauthorized access to equipment in remote shutdown stations (4.9.1)

P.10.V-25 computer room within the MCR security boundary (4.9.1)

P.10.V-26 design of emergency operations facility (EOF) (4.9.4)

P.10.V-27 modification to security boundaries during an emergency (4.9.4)

P.10.V-28 data storage methods (4.9.4)

P.10.V-29 compliance of perimeter intrusion alarm system with 10 CFR 73.55(h) (5.2.1 and 5.2.5)

P.10.V-30 signal transport delay (5.2.5)

P.10.V-31 analog-to-digital and digital-to-analog converters (5.7)

P.10.V-32 software requirement specification (6.1.2 and 6.1.6)

P.10.V-33 verification of software (6.1.2)

P.10.V-34 documentation of testing and verification of commercially available software (6.1.2)

P.10.V-35 configuration control of software purchased through software clearinghouses (6.1.2)

P.10.V-36 specific methods used to meet the requirement for diversity (6.1.6)

P.10.V-37 elimination of EMI (6.2.2)

P.10.V-38 signal validation methodology (6.2.2)

P.10.V-39 restoration state of control system components after loss of power (6.3.2)

P.10.V-40 setting resolution for control parameters (6.3.3)

P.10.V-41 neutron monitoring M-MIS (7.4)

P.10.V-42 selection of variables for automatic actuation (8.2.3)

P.10.V-43 criteria for area radiation monitoring system (10.2.1)

P.10.V-44 compliance with Item II.F.1.3 of NUREG-0737 (10.2.1)

P.10.V-45 radiation monitor placement, calibration frequency, and emergency power provisions (10.2.1)

P.10.V-46 operating philosophy (2 of Appendix B)

P.10.V-47 use of mockups, prototypes, and simulators (2 of Appendix B)

The following vendor- or utility-specific items remain and will be addressed during the staff's review of an individual application for FDA/DC. The staff's position on each item is provided in the appropriate section of this chapter, as indicated by the parenthetical notation following each item.

P.10.V-1 graded classification system (1.4.2)

P.10.V-2 defense-in-depth requirements for M-MIS (1.4.2, 4.4 and 4.5)

P.10.V-3 building confidence in the software (1.4.2)

P.10.V-4 separation of data transmission cables (1.4.2 and 6.2.11)

P.10.V-5 independent and diverse control equipment (1.4.2)

P.10.V-6 independent and diverse display equipment (1.4.2)

P.10.V-7 safety classification system (1.4.2)

P.10.V-8 reliability and maintainability analyses (1.4.2 and 3.5.4)

P.10.V-9 protection from operator error (2.3)

P.10.V-10 review of equipment used for displays to the operator (2.3)

P.10.V-11 methods to ensure operator alertness (2.3)

P.10.V-12 additional criteria for developing technology (2.3)

P.10.V-13 use of commercial-grade equipment (3.1.2)

P.10.V-14 complexity of M-MIS (3.1.3)

P.10.V-15 use of unproven technology (3.2.2)

P.10.V-16 quantitative reliability criteria (3.5)

P.10.V-17 selection of equipment failure modes (3.5.1)

P.10.V-18 maintenance frequency (3.5.2)

P.10.V-19 automatic reconfiguration after failure detection (3.6.4)

P.10.V-20 surveillance period of automatic testing features (3.6.8)

P.10.V-21 automatic bypass initiation (3.6.10, 3.6.12, and 3.6.14)

P.10.V-22 module software concerns (3.7.4)

P.10.V-23 bypass and test lockouts during on-line repairs (3.7.6)

P.10.V-24 MCR staffing (4.2)

P.10.V-25 alarm suppression techniques (4.3.3)

P.10.V-26 use of key-locked controls (4.5)

P.10.V-27 frequency allocation plan for communication systems (4.6)

P.10.V-28 use of "dial-up" telephone-type portable radios for security purposes (4.6)

P.10.V-29 unauthorized access to equipment in remote shutdown stations (4.9.1)

P.10.V-30 computer room within the MCR security boundary (4.9.1)

P.10.V-31 design of emergency operation facility (4.9.4)

P.10.V-32 modification to security boundaries during an emergency (4.9.4)

P.10.V-33 data storage methods (4.9.4)

P.10.V-34 special physical security measures for data transmission cable (5.2.1)

P.10.V-35 compliance of perimeter intrusion alarm system with 10 CFR 73.55(h) (5.2.1)

P.10.V-36 signal transport delay (5.2.5)

P.10.V-37 analog-to-digital and digital-to-analog converters (5.7)

P.10.V-38 software requirement specification (6.1.2)

P.10.V-39 quality assurance requirements for safety-related software (6.1.2)

P.10.V-40 verification of software (6.1.2)

P.10.V-41 documentation of testing and verification of commercially available software (6.1.2)

P.10.V-42 configuration control of software purchased through software clearinghouses (6.1.2)

P.10.V-43 definition of reasonable testing and degree of confidence to be considered (6.1.5)

P.10.V-44 specification of the level of diversity in safety systems (6.1.6)

P.10.V-45 guidance on performance of reliability evaluation (6.1.6)

P.10.V-46 specific methods used to meet the requirement for diversity (6.1.6)

P.10.V-47 elimination of EMI (6.2.2)

P.10.V-48 signal validation methodology (6.2.2)

P.10.V-49 failure modes selection (6.2.7)

P.10.V-50 restoration state of control system components after loss of power (6.3.2)

P.10.V-51 setting resolution for control parameters (6.3.3)

P.10.V-52 neutron monitoring M-MIS (7.4)

P.10.V-53 alternate means of inventory monitoring (7.13)

P.10.V-54 selection of variables for automatic actuation (8.2.3)

P.10.V-55 criteria for area radiation monitoring system (10.2.1)

P.10.V-56 compliance with Item II.F.1.3 of NUREG-0737 (10.2.1)

P.10.V-57 radiation monitor placement, calibration frequency, and emergency power provisions (10.2.1)

P.10.V-58 operating philosophy (2 of Appendix B)

P.10.V-59 use of mockups, prototypes, and simulators (2 of Appendix B)

P.10.V-60 inclusion of five elements in human factors organizational structure (3.1.2 of Appendix B)

P.10.V-61 scheduling of human factors studies and evaluation (3.1.3 of Appendix B)

P.10.V-62 documentation of technical bases for implementation of M-MIS process (3.2.1, 3.2.2, and 3.2.3 of Appendix B)

P.10.V-63 configuration of operator's workstation (3.2.5 of Appendix B)

P.10.V-64 illumination levels (3.3 of Appendix B)

P.10.V-65 use of IEEE P1023/D5 (3.7.6 of Appendix B)

2 SCOPE, OBJECTIVES, AND POLICY STATEMENTS

2.1 Scope

Section 1 of Chapter 10 of the passive Requirements Document states that each M-MIS will include the following equipment:

- data-gathering equipment that monitors equipment and process variables
- data communication equipment that transmits equipment and process variables between data processing equipment and plant equipment
- data processing equipment that manipulates data for use by plant personnel and/or automatic protection and control equipment
- plant information display and control equipment that provides alarm and display media for plant personnel to access plant processes and equipment status, and controls to operate plant equipment
- output processing equipment that provides the necessary interface between plant controls and plant equipment actuators

As described in the other chapters of the passive Requirements Document, the interface between the fluid systems and the instrumentation system is considered to be at the connection of the fluid system to the sensor. For control systems, the interface is the connection to the final actuator wiring or tubing through which the power or control signals will be transmitted. Chapter 10 of the passive Requirements Document provides design criteria that incorporate technological improvements and human factors considerations that will resolve many problems experienced by operating reactors. The staff concludes that EPRI's definition of the M-MIS and the goals are acceptable.

Section 1 of Chapter 10 states that the scope includes all M-MIS equipment regardless of supplying organization. Although the requirements are not all inclusive, they are aimed at resolving specific problem areas or meeting specific utility objectives. The M-MIS requirements include:

- instrumentation, including sensors and local instruments, for all safety and non-safety systems throughout the plant
- automatic and manual controls for all safety and non-safety systems
- protection functions, including safety and non-safety systems
- diagnostic systems, including loose parts monitoring systems, rotating machinery diagnostics, and neutron noise monitoring
- monitoring and control stations for the plant systems, including the MCR, remote shutdown control station, technical support center (TSC), EOF, and local control stations (Not every local control panel is specified, but requirements are provided regarding when local controls should be provided and for the consolidation and arrangement of these local controls into panels.)

- I&C power supplies, grounding, and environmental compatibility
- computer systems for control, data acquisition, display, storage and retrieval, monitoring and alarms, technical support, and operations support
- plant communications systems, including data, visual, and voice intra-plant communication associated with plant operation and maintenance
- a simulator for the design, verification, and validation of the M-MIS, but not the simulator complex itself

2.2 Objectives

Section 2.1.1 of Chapter 10 of the passive Requirements Document requires that the M-MIS design take full advantage of operator capabilities without challenging operator limitations and that the human component be specifically included in the design.

Section 2.1.2 of Chapter 10 states that the M-MIS design will be coordinated with the overall plant design. The staff agrees with this objective.

Section 2.1.3 of Chapter 10 states that the control, protection, and monitoring functions of the M-MIS will be designed with a consistent, integrated approach so that they work together to enhance plant operation and reduce operator burden. All control stations will be integrated so that they perform as a coordinated whole and provide a consistent and easily understood and manipulated man-machine interface for plant operations and maintenance personnel. The staff agrees with this approach; however, consideration of such requirements as segmentation, separation, and single-failure criteria needs to be implemented for the integrated control stations. This issue is discussed further in Section 2.3 of this report under the heading "Main Control Room."

Section 2.1.4 of Chapter 10 states that the M-MIS will be constructed of highly reliable components and equipment, be well analyzed, and be tested; however, the specific analyses and tests to be performed are not discussed. The M-MIS will be designed so that failures or problems in one function will not be able to propagate into other functions to minimize the extent of the upset and not increase operator burden. The staff agrees with this requirement. The staff's evaluation of the EPRI reliability requirements are included in Section 3.5 of this report.

Section 2.1.5 of Chapter 10 specifies that maintenance and possible future upgrades will be considered in the design of the equipment. The staff considers maintenance and upgrades to be areas that need improvement, compared to current operating reactors, and therefore, agrees with this requirement.

2.3 Policy Statements

EPRI intends the policy statements in Section 2.2 of Chapter 10 of the passive Requirements Document to provide a summary view of the direction to be taken for the requirements. These are discussed below.

M-MIS Systems Approach

Section 2.2.1 of Chapter 10 requires that the M-MIS employ modern digital technology and that it be robust, segmented, fault tolerant, and highly reliable. Redundant equipment within a segment will be separated, and protection against the propagation of failures will be provided. Some parameters will have signal validation. Multiplexing and fiber optics will be used when appropriate. EPRI requires that hardware and software be standardized and modularized. The equipment will include self-diagnostics and will be compatible with the environment in which it is located. The staff concludes that these requirements are consistent with engineering practices that result in high-quality, digital-based I&C systems and are acceptable.

Design Process

Section 2.2.2 of Chapter 10 states that the M-MIS design process is to be directed by a single organization and will include utility engineering, operations, and maintenance personnel. Human factors considerations will be integrated into the design effort. A continuous V&V effort will be performed parallel to the design by an independent team with particular emphasis on software design. The staff finds this policy acceptable.

Reliability Inherent in Design

Section 2.2.3 of Chapter 10 states that the M-MIS will possess defense against the propagation of faults through segmentation, independence, and other measures. These systems also will be sufficiently robust to prevent a single failure from causing a forced outage. In its request for additional information (RAI) dated April 10, 1990, the staff asked EPRI to describe the other measures to be used. The staff concluded that this requirement might be too vague to provide sufficient guidance for a designer to use effectively. The staff further noted that this requirement appeared to preclude an integrated control system (ICS) design. It asked EPRI to address the tradeoffs between integrated and segmented designs.

By letter dated July 23, 1990, EPRI responded that the main purpose of segmentation is to prevent the propagation of failures. The statements in the passive Requirements Document are meant to encourage a vendor to take whatever measures are needed to attain the goal. EPRI further stated that the segmentation requirement would prohibit a Babcock and Wilcox (B&W) ICS in ALWR plants. It stated that a highly interconnected, ICS was not necessary to achieve adequate plant performance and was more vulnerable to problems from a single failure.

The staff concludes that these requirements are acceptable. However, it does not preclude the use of a B&W ICS system. Specific reliability issues are evaluated in Section 3.5 of this chapter.

Testing of Man-Machine Interface Systems

Section 2.2.4 of Chapter 10 states that the equipment used in the M-MIS will be designed to support inservice testing and specifically to avoid the use of lifted leads and jumpers. The staff considers this to be good practice and, therefore, acceptable.

Proven Technology

In Section 2.2.5 of Chapter 10, EPRI encourages the use of modern technology to solve existing problems and make significant improvements over previous designs. However, it plans to use equipment and associated software/firmware with extensive experience or testing to avoid introducing new problems. The staff concludes that this policy is acceptable.

Operating Staff

Section 2.2.7 of Chapter 10 gives the number of people that the M-MIS design must accommodate. The designer will verify the operator staffing levels using mockups and dynamic simulation. EPRI anticipates that one reactor operator will be able to control the plant during normal power operation. Additional information on operating staff is provided in Section 4.2 of Chapter 10 of the passive Requirement Document. The staff addresses the human factors aspect of this issue addressed in Appendix B of this chapter.

In its RAI dated April 10, 1990, the staff asked if expert systems or "knowledge-based" software are allowed or required to reduce operator burden and what the criteria would be for their use. The use of systems that might be considered expert systems is allowed in the passive Requirements Document. In its response dated July 23, 1990, EPRI stated that the criteria pertaining to need and demonstrated capabilities required for the M-MIS will apply to any expert systems. This does not violate any existing NRC criteria and is, therefore, acceptable. However, computer systems that are commonly referred to as "expert systems" or "artificial intelligence systems" will only be acceptable to the staff for use in safety systems when they can be shown to be highly reliable and specific design and acceptance criteria are established.

Human Factors Engineering

Section 2.2.8 of Chapter 10 states that significant emphasis will be placed on human factors during the M-MIS design process. Particular emphasis will be placed on

- elimination of potential sources of human error
- reduction in the probability of human error through careful selection and allocation of tasks
- provision for the detection of and recovery from human errors should they occur and the detection of errors before they affect the plant

The staff agrees that human factors should be considered during the design process rather than after. Appendix B of this chapter contains an evaluation of this subject. In its RAI dated April 10, 1990, the staff asked if the elimination of potential sources of human error meant that the I&C was to be single-operator error proof. The staff also asked EPRI to explain how the goal of error detection before the errors affect the plant is to be reconciled with a reduction in anticipatory trips and simplification.

By letter dated July 23, 1990, EPRI responded that it is probably not possible to make a practical M-MIS that will be single-operator error proof, unless the systems are totally automatic. It considers operator error to be a factor that the designer must address. The primary method in the passive Requirements Document to reduce operator error is the basic improvement in human factors considerations during the design. EPRI further stated that it considers simplification to be consistent with reduction in operator errors because of ease of use. Although the staff agrees with the EPRI comment, it is aware that some vendors incorporate protection within their software programs to prevent incorrect operator actions such as exceeding a setpoint. The descriptions of these internal protection methods by the vendors in the passive Requirements Document add additional complexity to the system. The benefits of these features as well as other complexities (self-diagnostics) may provide significant safety gains. Therefore, these requirements are acceptable; however, the staff does not consider that such protection simplifies the system. Therefore, it will address the tradeoffs associated with the complexity of such software during its review of an individual application for FDA/DC.

Level of Automation

Section 2.2.9 of Chapter 10 states that the designer will consider the system response requirements, complexity of operation, operator burden, level and duration of attention required, and so forth, when determining the proper level of automation. Operator awareness of plant status and operator alertness also must be considered. Automatic control systems should use fail-safe features when possible. The staff does not disagree with these policies and, therefore, finds them acceptable; however, it will review each specific design to assess the level of automation.

Main Control Room

Section 2.2.10 of Chapter 10 states that the MCR will be designed by an interdisciplinary control room design team that is a subset of the M-MIS design team. The MCR equipment and information will be integrated and coordinated with other plant functions, such as engineering, maintenance, management, and emergency response facilities. Personnel in the MCR will use redundant compact workstations with multiple, electronic display and control devices that provide organized, hierarchical access to alarms, displays, and controls. Large, upright integrated plant status panels and top-level alarm displays viewable from anywhere in the MCR also will be used. Appendix B to this chapter contains the staff's human factors assessment of the passive Requirements Document MCR design process.

EPRI requires the designer to consider the use of multifunctional display and control features. An example provided by EPRI includes the use of a multifunctional control device for the operation of redundant safety trains and of both safety and non-safety equipment.

The passive Requirements Document requires that workstations be designed to satisfy existing regulations such as Institute of Electrical and Electronics Engineers (IEEE) 279, "Criteria for Protection Systems for Nuclear Power Generating Stations"; IEEE 384; and Regulatory Guide (RG) 1.62, "Manual Initiation of Protective Actions." RG 1.62 stipulates that no single failure

within the manual, automatic, or common portion of the protection systems should prevent initiation of protective action by manual or automatic means. The designer will be required to use existing defensive measures as appropriate to ensure that the alarm, display, and control functions provided by the redundant workstations meet these standards. Methods given in the passive Requirements Document to meet these requirements include segmentation, separation, independence, diversity, fault tolerance, signal validation, self-testing, error checking, and supervisory watchdog programs.

EPRI requires that a supervisor's workstation identical to the operator's workstation be provided. Each workstation is required to provide the following features:

- electronically displayed normal, abnormal, and emergency operating procedures
- electronically displayed presentations of plant operational parameters and technical data based on operator tasks and event categories
- electronically displayed piping and instrument diagrams logically organized for easy access and dynamically updated with equipment status information
- electronically displayed alarms designed to minimize nuisance alarms, and diagnostic aids
- controls that are coordinated with the decisionmaking approach; the controls do not need to be spatially dedicated

A series of integrated upright displays and alarms that are redundant will be placed on the MCR walls to provide a large plant mimic that will contain only a few key parameters.

In its RAI dated April 10, 1990, the staff stated using a multifunctional control device for the operation of redundant safety trains and of both safety and non-safety equipment appeared to be a basic violation of all standards involving diversity, redundancy, separation, and defense-in-depth. By letter dated July 23, 1990, EPRI agreed that a multifunctional device could be constructed that would violate these requirements, but that the design approaches it intends to use would meet the requirements while providing a desirable compact workstation.

In its response, EPRI stated that the workstations will be redundant with any single station capable of providing all needed monitoring and control functions. In addition, Section 4.5.6 of Chapter 10 of the passive Requirements Document requires that all automatically actuated safety functions will have manual, conventional, hardwired system-level actuation controls in the MCR. Information indicating that the systems have been actuated also will be provided independent of the workstations.

Section 2.2.10 of Chapter 10 states that redundancy in the workstations will be provided so that the inoperability of any one workstation does not prevent

the operators from sharing their tasks between two operators at different workstations. Certain controls important to safety will be provided in a separate and diverse manner.

In addition, Section 2.2.10 provides alarm requirements that are consistent with those stated in the policy issue on control room annunciator reliability in SECY-93-087.

In its RAI dated April 17, 1991, the staff requested more information on the development of computerized procedures. By letter dated June 13, 1991, EPRI provided the rationale for and advantages of electronically displayed procedures. The staff concludes that the development of electronically displayed procedures is a desirable goal. However, they must be considered early in the design process. If electronically displayed procedures are determined to be an improvement, they should be provided as part of the design.

In its RAI dated April 10, 1990, the staff asked EPRI to define the phrase "electronically displayed." In addition, the staff asked why the top-level key parameter display for essential equipment is required not to be redundant. The staff considers that little guidance was provided concerning the parameters that are to be displayed. By letter dated July 23, 1990, EPRI responded that electronically displayed is intended to encompass cathode ray tubes, plasma displays, and liquid crystal displays. Specifically mentioned as items not included were meters, strip chart recorders, annunciator windows, and backlit indicators. EPRI stated that there was no need to make the overview board redundant because the information presented would also be available at the workstations. The human factors aspect of the displays is addressed in Appendix B to this chapter. The staff will review the hardware and software aspects of the design within the human factors and I&C scope of review for the specific equipment selected during its review of an individual application for FDA/DC.

Section 2.2.10 describes measures to be taken to enhance operator alertness. In its RAI dated April 10, 1990, the staff asked EPRI to provide additional information on how this was to be accomplished. In the letter dated July 23, 1990, EPRI referenced Section 3.4.2 of Chapter 10 of the passive Requirements Document and noted that further research was being conducted in this area. EPRI considers some methods used to maintain operator alertness to be beyond the scope of the passive Requirements Document. The staff will address this issue during its review of an individual application for FDA/DC.

Other Control and Monitoring Stations

Section 2.2.11 of Chapter 10 states that the same design policies as those used for the MCR M-MIS will be used for the TSC, EOF, remote shutdown station, local control stations, and other monitoring facilities. The passive Requirements Document states that the remote shutdown station will rely on local operation of some selected equipment.

In its RAI dated April 10, 1990, the staff asked EPRI to explain why some equipment would be excluded from the remote shutdown panel and what the exclusion criteria are. By letter dated July 23, 1990, EPRI responded that this section of the passive Requirements Document provides only an overview of

the remote shutdown characteristics and that the task analysis and requirements described in Sections 3.1.3.3.3 and 4.9.3.1 of Chapter 10 provide additional detail.

Protection From Obsolescence

Section 2.2.12 of Chapter 10 notes that state-of-the-art I&C is constantly changing. The M-MIS will be modularized and standardized to alleviate the problem of obsolescence of the equipment. The staff concludes that this is acceptable.

Regulatory Stabilization

Section 2.2.13 of Chapter 10 states that even though EPRI expects the M-MIS to meet the existing regulatory requirements, many of those requirements do not consider the use of modern technology. EPRI considers that the evolution of the detailed M-MIS design may necessitate new interpretation of the intent and purpose of existing guides and standards.

The staff agrees with the comments in this section. Although some new standards and RGs (e.g., RG 1.152, "Criteria for Programmable Digital Computer System Software in Safety-Related Systems of Nuclear Power Plants") that account for modern digital technology have been created, the standards and RGs in this area as a whole are still catching up with the technology. Until such clear standards are in place, the staff expects that issues relevant to determining the safety of a plant that uses modern technology will be resolved through extensive discussions and reviews.

The staff also expects the M-MIS to be designed to meet the requirements of RG 1.153, "Criteria for Power, Instrumentation, and Control Positions of Safety Systems," which endorses IEEE 603-1980, "Trial-Use Standard Criteria for Safety Systems for Nuclear Generating Stations." The staff will use additional criteria as needed during its review of an individual application for FDA/DC.

2.4 Conclusions

The staff concludes, with resolution of the unresolved vendor- or utility-specific items noted above, that the requirements in Section 2 of Chapter 10 of the passive Requirements Document do not conflict with current regulatory requirements and are, therefore, acceptable. However, the requirements alone do not provide sufficient information to make a determination that a specific design application will be acceptable. Therefore, applicants referencing the passive Requirements Document will be required to demonstrate compliance with the guidance in the SRP or provide justification for alternative means of implementing the associated regulatory requirements. In addition, the applicants will be required to adequately resolve issues unique to the use of digital technology.

3 KEY REQUIREMENTS

3.1 M-MIS Design Process Requirements

Section 3.1 of Chapter 10 of the passive Requirements Document states that the M-MIS design will be established by a defined process that is to be systematic and consistent. The following sections provide greater detail on this requirement. The staff agrees with the general criterion to keep the design of the M-MIS under the control of a formal design process.

3.1.1 Overall Design Process Requirements

Section 3.1.1 of Chapter 10 requires that the M-MIS design meet the functional requirements without unnecessary complexity and without the use of unproven or highly developmental instrumentation or control strategies. The staff agrees with these requirements.

EPRI states that the M-MIS will be designed to emphasize the functional division of the plant rather than division according to physical systems traditionally used. EPRI considers the basic functions for the overall control of the reactor to be reactivity control, reactor coolant pressure control, reactor coolant inventory and chemistry control, reactor core heat removal, and steam generator water level control (in pressurized-water reactors (PWRs)). EPRI states that for other sections of the plant, the functional divisions will be control of energy flow, control of the local plant environment, control of the release of material to the environment, and provision of services of water, air, and electric power.

The emphasis on functional design before system and equipment design is acceptable to the staff as long as the existing requirements are met. Some requirements, such as V&V and quality assurance (QA), need early implementation.

Section 3.1.1.2 of Chapter 10 states that the M-MIS design process will be fully integrated with the design processes of the remainder of the plant and that all modes of operation, including severe accidents, will be considered in the design. The staff agrees that some examples of integration of M-MIS design with the design process (insufficient HVAC) have shown that improvement in this area can be made.

EPRI states that the M-MIS design process will be applied consistently to all the interfaces between the plant and its operators and support staff throughout the plant, regardless of system, location, or safety classification. The staff agrees that a consistent application of design criteria is desirable. It addresses the human factors aspects of this section in Appendix B to this chapter.

3.1.2 M-MIS Design Organization and Plan

Section 3.1.2 of Chapter 10 provides criteria for the M-MIS design organization and plan. It specifies that the design will be formally organized and directed by a single organization. Also, the designer must have broad experience in systems design and operation as well as I&C design. EPRI states

that although it is recognized that the work will be carried out by a team, for simplicity, many of the requirements are stated as requirements on the "M-MIS designer," since this is the individual or group that will ultimately be responsible.

Section 3.1.2.2 of Chapter 10 states that at least one individual design group will be created within the M-MIS design group for the MCR and should include the following:

- engineers and designers for the nuclear steam supply system
- engineers and designers for the balance-of-plant systems
- human factors specialists
- utility engineering, operations, and maintenance staff

To achieve standardization, the staff considers that the inclusion of the utility in the design will require consensus between the various utilities on design, operation, and maintenance; this consensus does not currently exist.

Section 3.1.2.3 of Chapter 10 states that the design process will include independent review of all aspects of the M-MIS design. V&V of the M-MIS design will be performed. The staff finds this acceptable. However, details regarding the independence of the review team are not given. This section also references Nuclear Safety Analysis Center (NSAC)-39, "Verification and Validation for Safety Parameter Display Systems," which allows several different methods for accomplishing review team independence. One acceptable approach in NSAC-39 is for the utility to perform the review. The staff notes that because the utility is required in the passive Requirements Document to be part of the design team, care would need to be taken to ensure that the utility had separate groups with this capability. The second acceptable approach is to have a separate V&V contract with a firm experienced in V&V. A third acceptable approach in NSAC-39 is for the organization responsible for the development to perform or subcontract the V&V activities. However, individuals who do the V&V must not participate in the design or implementation of the M-MIS.

In the DSER for Chapter 10, the staff stated that the guidance in NSAC-39 was intended for the non-Class 1E safety parameter display systems. It stated that the approaches in NSAC-39 referenced as examples in the passive Requirements Document were acceptable provided additional clarification was added that, for safety systems, the verifier will be responsible to an organizational supervisor other than the designer. The independence of the V&V organization is very important and will provide a higher level of assurance that the verifier is sufficiently removed from the schedule and budget constraints of the design process and will remain independent. This was identified as an open issue. In addition, the staff will review details of the QA program of the designer to ensure that sufficient independence of each review team is maintained. In Revision 3, EPRI revised Section 3.1.4 and its rationale to require the M-MIS design review team to be technically, managerially, and financially independent of the design team. This revision clarifies the independence of the software V&V review teams. Therefore, this open issue is closed.

Section 3.1.2.4 of Chapter 10 states that the designer will prepare a comprehensive plan for the development and implementation of the M-MIS design that includes at least the following:

- schedule with milestones
- configuration control plan for the design and design tools
- plan for developing and qualifying tools
- M-MIS/plant systems interaction plan
- provisions for utility participation

In a letter dated July 23, 1990, EPRI stated that requiring designers to provide compilers for the application software would probably result in the designer developing a compiler for this special use, rather than using an existing, well-proven compiler. In the DSER for Chapter 10, the staff agreed that using a compiler with extensive commercial use might produce more reliable software than a special-use compiler and stated that EPRI should include this clarification in the passive Requirements Document. This was identified as an open issue. In Revision 3, EPRI revised the rationale in Section 3.1.2.4 to state that the plant designer must comply with the design process of Section 6.1.2 and the software design requirements of Section 6.1.3. This revision clarifies the requirements for using commercial compilers for software in safety systems. Therefore, this open issue is closed.

Dedication of commercial-grade equipment for use in safety systems has been an area of concern to the staff. Problems have included processing methods and vendors who misrepresent the capabilities or qualification of their equipment. A more comprehensive discussion of past concerns and actions can be found in NRC Generic Letter (GL) 89-02, "Actions To Improve the Detection of Counterfeit and Fraudulently Marketed Products," and GL 91-05, "Licensee Commercial-Grade Procurement and Dedication Programs." These problems have been experienced with equipment that has significantly longer operating history than the microprocessors and other newer equipment used in the M-MIS. In its RAI dated April 10, 1990, the staff asked EPRI to describe the method to be used to determine if past design work on commercial-grade equipment was done properly and could be used for nuclear safety system applications to take advantage of proven designs. The staff also asked if reverse engineering (determination of requirements from the end product) was an appropriate method for software development. By letter dated July 23, 1990, EPRI responded that the intent of the requirement pertaining to proven design is to discourage the use of components that have a short operational history.

The staff concludes that this is a good design philosophy, but it will need to evaluate design-specific implementation for specific products during its review of an individual application for FDA/DC.

The staff believes that it is possible to establish a realistic threshold of experience with the operation of software to substitute for documented V&V. However, it believes that this threshold is difficult to determine without a specific design to evaluate. No program is in place that would ensure that the end users and the staff are apprised of any problems caused by changes to commercial-grade software tools and other functional software.

None of the requirements in Section 3.1.2 are in violation of NRC requirements and are, therefore, acceptable. However, additional requirements are needed for dedication of commercial-grade equipment for use in safety systems. In its RAI dated May 17, 1991, the staff asked EPRI to describe the methodology for dedicating commercial software. In its response dated July 22, 1991, EPRI stated that dedicating commercial software for use in the design and for use as design tools was the M-MIS designer's responsibility and referred to the requirements in Sections 6.1.2, 6.1.3, and 6.1.5 of Chapter 10 of the passive Requirements Document regarding acceptance testing of software, software design, development, testing, and documentation of tools and techniques. EPRI has issued NP-5652, "Guideline for the Utilization of Commercial Grade Items in Nuclear Safety related Applications (NCIG-07)," but this document does not specifically address software or tools. On the basis of EPRI's activities in the commercial dedication arena, the staff concluded in the DSER for Chapter 10 that EPRI should provide additional guidelines for the dedication of such equipment. This was identified as an open issue. In Revision 2, EPRI revised Section 3.2.1 to refer to the guidance in NP-5652. EPRI explained that there are no industry guidelines for the dedication of commercial-grade computer equipment or software; however, EPRI, utilities, and others have embarked on a program to develop the qualification of commercial-grade programmable logic controls and the dedication of commercial-grade. The staff agrees with EPRI's actions. Therefore, this open issue is closed.

3.1.3 Required Design Process Features

Section 3.1.3 of Chapter 10 states that the M-MIS design process will ensure that problems with existing M-MIS designs are identified and features are incorporated in the ALWR M-MIS that provide satisfactory solutions to these problems.

The designer will include the following in the design process:

- a comprehensive and systematic review at the beginning of the design process of existing LWR M-MIS designs to identify problems that have led to safety concerns, low plant availability, and high maintenance burdens
- an assessment of each identified problem for its applicability to the ALWR
- identification in the final M-MIS design of how each of the problems has been solved
- independent reviews of the identification, assessment, and solutions of the problems

In its RAI dated May 17, 1991, the staff requested a list of the sections that are affected by the resolution of the design problems encountered by the French N4 plant, the Canadian Darlington A plant, and the Sequoyah Eagle 21 system. These events clearly indicated to the staff a failure to impose a rigorous, formal, structured V&V program from the very start of the design. In its response of July 22, 1991, EPRI referred to Section 3.1.3.1 regarding requirements for the resolution of past problems, to Sections 3.1.1 and 3.1.4

regarding simplicity of system design, to Section 6.1.3 regarding requirements on software complexity, and to Sections 6.1.2 and 6.1.5 regarding requirements on specifications of software and on V&V.

The staff concludes that the requirements of this section should help ensure that problems with existing M-MIS designs are identified and avoided in future ALWR designs and are, therefore, acceptable. The staff's evaluation of the human factors aspects of this section is provided in Appendix B to this chapter.

Section 3.1.3.2 of Chapter 10 requires that the following actions be taken in the design process to achieve a simple, plantwide standardized design:

- preparation of guidelines for standard design practice
- development and use of standard component and systems designations
- tracking of the numbers and types of components

In its RAI dated April 10, 1990, the staff asked EPRI to show that the simplification and standardization of the plant do not compromise the defense-in-depth principles of General Design Criterion 22 of 10 CFR Part 50, Appendix A. The staff was concerned that equipment standardization within a single plant could reduce the diversity that exists in past designs.

By letter dated July 23, 1990, EPRI responded that there was no inference that safety should be compromised to achieve its goals and described the positive aspects of simplification, standardization, and segmentation. However, the staff does not believe that the extensive use of computer-based systems is a simplification in the area of I&C. EPRI also noted that the advanced technology planned for the ALWR M-MIS will be more complex than current systems when measured by numbers of components. There is extensive experience, in particular with foreign reactor projects, that demonstrates that these new designs are more complex than past designs and require significant rigor in design control, such as strictly implemented V&V. The staff does not discourage the use of computer-based systems because there does appear to be a real potential for increases in safety for the reasons given throughout Chapter 10 of the passive Requirements Document. The staff will evaluate the acceptability of the level of complexity of the system design during its review of an individual application for FDA/DC.

The staff notes that the standardization referred to in this chapter of the passive Requirements Document refers to the use of standard equipment within a single plant. This chapter does not address the 10 CFR Part 52 issue of a standard design that would be a single total design implemented at more than one site. The staff concludes that the passive Requirements Document does not necessarily require future plants to be of the same design in the M-MIS area, although there are no requirements specifically precluding standard plant designs.

In its RAI information dated April 10, 1990, the staff stated that design standards and practices should be established before the beginning of the design process. Section 3.1.3.2 of Chapter 10 of the passive Requirements Document states that the design process will provide for the preparation of guidelines for standard design practices. In its letter dated July 23, 1991,

EPRI defined "design practices" to be detailed rules and not general guidelines for design. EPRI further stated that use of new technology will require the testing and evaluation of many alternatives before the design practices are selected.

Section 3.1.3.3 of Chapter 10 states that the M-MIS design process will result in the explicit identification of the functions needed and the individual tasks needed to perform the functions. The functions and tasks will include all of those that affect the design and will be analyzed and validated using methods and equipment including mockups, modeling, and a full-scale simulator.

The following list is a minimum set of functions identified in Section 3.1.3.3.3 of Chapter 10 that will be provided:

- allocation of functions between automatic and manual control
- allocation of tasks among workstations
- development of control and operation strategies
- assignment of responsibilities of the operating crew
- assessment of operator mental and physical workload
- selection of types of displays and their detailed characteristics
- selection of controls and their detailed characteristics
- selection and arrangement of alarms and their integration into the control station designs
- development of operating procedures and training requirements
- evaluation of the effects of credible M-MIS equipment failures
- V&V reviews
- arrangement of workstations

The identified tasks and functions, including the bases for the allocations, will be documented.

Section 3.1.3.4 of Chapter 10 states that the potential for and the consequences of failures of plant and M-MIS components will be explicitly considered in the M-MIS design process. The functions and tasks required of the operator when equipment fails will be identified. The analysis and validation testing of the M-MIS will include the failures and recovery from them. In the DSER for Chapter 10, the staff concluded that the clear indication of such failures and the effects of any cascading or interaction failures also should be considered and this requirement should be added. This was identified as an open issue. By letter dated September 29, 1992, EPRI explained that Sections 3.1.3.3.3 and 3.1.3.4 require the M-MIS design process to explicitly consider the failures of components. These sections also require the functions and tasks that result from the operator coping with equipment failures

to be identified as part of the M-MIS design bases. This explanation clarifies the requirements for analysis and validation testing of M-MIS. Therefore, this open issue is closed.

In response to a staff concern about consideration of rapid prototyping in the M-MIS design process to help ensure the completeness of the system specification and the testability of the design concept, in Revision 4, EPRI added Section 3.1.3.5.1, which requires rapid prototyping as part of the M-MIS design process.

Section 3.1.3.5.2 of Chapter 10 states that the design process will include the development of digital computer-based dynamic models for the overall plant response as well as individual control systems. These dynamic models will be

- suitable for analyzing both steady-state and transient behavior
- used to confirm the adequacy of control schemes
- used to confirm the allocation of control to an automatic system or an operator
- used to develop and validate plant operating procedures
- validated against tests of actual plant behavior, wherever practicable
- developed early enough in the design process so that the systems can be modified if necessary
- incorporated into the simulators
- documented

In its RAI dated April 10, 1990, the staff asked EPRI to supply information concerning dynamic models and validation testing. By letter dated July 23, 1990, EPRI stated that the dynamic models will receive the same level of V&V as other design programs. EPRI's statement that the validation of the models against actual plant behavior is to be performed "whenever practical" is intended only to eliminate tests that cannot be performed at the actual plant, such as under severely degraded conditions. The staff expects that confirmatory measurements during startup to validate software algorithms will be required, and EPRI indicates that it does not intend to eliminate this validation step. The staff concludes that this is acceptable. The staff's evaluation of the human factors aspects of this section is provided in Appendix B to this chapter.

In the DSER for Chapter 10, the staff stated that EPRI should consider an additional requirement in the M-MIS design process for the use of rapid prototyping in the development and validation of the functional specifications. This was identified as an open issue and also was discussed in Part I of the DSER for Chapter 10. In Revision 4, EPRI added Section 3.1.3.5.1 to require that the M-MIS design process include the use of

rapid prototyping in the development and validation of functional specifications. This revision provides adequate requirements for the use of rapid prototyping in the development and validation of functional specifications. Therefore, this open issue is closed.

Section 3.1.3.5.3 of Chapter 10 states that the control systems will be analyzed to ensure stability and correct response. Delays in signal propagation will be considered.

Section 3.1.3.6.1 of Chapter 10 states that the design process will define the test requirements in formal test plans. The test plans will include the original validation testing, operational tests, and post-maintenance tests. As a minimum, each test plan will

- identify the items to be tested
- identify any features that are not to be tested and the reasons why
- describe the test approach
- specify the test case specifications
- specify the acceptance criteria
- specify the test environment
- specify the test equipment
- identify the group performing the test
- identify the test staffing requirements
- specify the test sequence and schedule

EPRI requires the designer to define and write the formal test plans. In its RAI dated April 10, 1990, the staff stated that most V&V and testing programs do not allow programmers to test their own software at the formal test stage, although the developer should be required to do testing during the development phase. Someone other than the developer should decide if the requirements and goals of the software have been met, although any testing that is recommended by the developer should be performed during formal testing. The staff agrees with the requirement that the developer review and correct the results of the testing. By letter dated July 23, 1990, EPRI responded that Section 6.1.5.7 of Chapter 10 of the passive Requirements Document requires the testing of the software to be independent of its development. The staff concludes that the requirement for independent testing is acceptable.

Section 3.1.3.6 of Chapter 10 states that any inservice surveillance testing required to ensure equipment operation or to meet NRC requirements will be included in the previously listed test plans. Installation and startup tests also will be included in these test plans. The staff's evaluation of the human factors aspects of this section is included in Appendix B to this chapter.

3.1.4 Independent Review of Design Process

Section 3.1.4 of Chapter 10 requires an independent review that includes V&V as well as an overview of the entire design process. As discussed in Section 3.1.2 of this chapter, the independence of the review team must be specified to be acceptable to the staff.

Section 3.1.4.1 of Chapter 10 states that the reviewers will have technical qualifications comparable to those of the designers. This is in accordance with American National Standards Institute/Institute of Electrical and Electronics Engineers (ANSI/IEEE) 7-4.3.2-1982, "Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations," and is therefore, acceptable to the staff.

EPRI requires that a preliminary review plan be established before the design process is initiated. The final review plan will be established jointly by the designers and the reviewers. The independent review will start when the design process is started and will continue through final testing.

Section 3.1.4.4 of Chapter 10 states that the reviewers will verify that the M-MIS and plant systems functional requirements are met and that the hardware and software specifications will satisfactorily implement the functional requirements. The reviews will address

- simplicity.
- standardization
- reliability and availability
- protection against common-mode failures
- power supply failures and their effects
- compatibility with the environment, including
    - temperature
    - humidity
    - radiation
    - radiofrequency interference
    - electromagnetic interference
    - vibration and seismic loadings
    - fire and fire suppression systems
    - flood
    - electrical transients and surges.
- maintainability
- human factors for operators and maintenance personnel
- protection against obsolescence
- flexibility and expandability
- constructibility
- testability

The staff concludes that consideration of these items in the design review is good engineering practice and is acceptable.

EPRI states that the review team will review the system and component test plans, witness the tests, and review the test results. The review team is required to review the completeness and accuracy of the design documentation. It also will document its reviews and any deficiencies discovered.

3.2 Proven Technology

Section 3.2 of Chapter 10 of the passive Requirements Document states that proven technology will be used except when not available. In those cases, the use of advanced systems, equipment, software, and firmware may be justified if proven in other applications.

3.2.1 Criteria for Proven Technology

Section 3.2.1 of Chapter 10 includes a reference to Section 11.2.1 of Chapter 1 of the passive Requirements Document, which defines proven technology as an item that has been used for several years in LWR plants or is otherwise specifically proven by the utility to be suitable for the application. Chapter 1 also allows the use of unproven technology, if justified. Section 3.2.1 of Chapter 10 states that in addition to the Chapter 1 definition, M-MIS equipment will be considered proven if (1) it has at least 3 years of documented, satisfactory service as modules of subsystems in power plant or non-power plant applications similar to that in LWRs or (2) it has undergone a defined program of prototype testing.

EPRI states that the primary emphasis of the ALWR is reliable power production. The staff concludes that EPRI's definition of proven technology is acceptable provided all the Commission's safety requirements are met.

In its RAI dated May 17, 1991, the staff questioned whether a design like the B&W ICS would be acceptable according to the requirements of this section. In its response dated July 22, 1991, EPRI stated that the ICS will not be used in the ALWR because it will not meet the requirements for segmentation (Section 3.5.3.1 of Chapter 10 of the passive Requirements Document) and the requirement of Section 3.2.1. However, EPRI noted the components used in the B&W ICS will meet the requirements of Section 3.2.1 and could be used in the ALWR application. The staff also requested EPRI to reference EPRI NP-5652, "Guideline for the Utilization of Commercial-Grade Items in Nuclear Safety-Related Applications (NCIG-07)." EPRI has made this reference in Section 3.2.1.

3.2.2 Criteria for Use of Unproven Technology

If a defined gain in simplicity or performance is needed that cannot be obtained with proven equipment, Section 3.2.2 of Chapter 10 allows the use of unproven equipment if testing or experience data justify it. In its RAI dated May 17, 1991, the staff questioned the role of prototype testing in the use of proven and unproven technology. In its response dated July 22, 1991, EPRI stated that Section 3.2.1 states that equipment can be considered "proven" if it has completed a defined program of prototype testing. Section 3.2.2 provides for the use of equipment that cannot be considered "proven" according to the criteria of Section 3.2.1. (Prototype testing itself is covered by test plans of Section 3.1.3.6.1 and subject to the independent review requirements of Section 3.1.4.4.4). The staff will evaluate the acceptability of the use of such equipment during its review of an individual application for FDA/DC.

3.3 Cost

Section 3.3 of Chapter 10 of the passive Requirements Document states that the costs of the M-MIS will be consistent with the same general cost guidance in Chapter 1 for other equipment, with an emphasis on total life cycle and plant availability considerations instead of simple initial cost considerations. Although cost considerations are outside the scope of its review, the staff considers that the emphasis on long-term reliability and maintenance considerations is a positive goal.

3.4 Operator Actions

Section 3.4 of Chapter 10 of the passive Requirements Document states that the operator and other plant personnel are to be considered in the M-MIS design.

3.4.1 Operator Workload and Vigilance

Section 3.4.1.1 of Chapter 10 states that the M-MIS design will not require the performance of tasks that would overburden the operators during normal, upset, or emergency conditions.

Section 3.4.1.2 of Chapter 10 states that the M-MIS will be designed with features that will facilitate operator activities and tend to keep them alert and attentive. The passive Requirements Document requires the application of the guidance provided by EPRI NP-6748, "Control-Room Operator Alertness and Performance in Nuclear Power Plants." In the rationale, EPRI notes that the industry is studying the issue of operator vigilance and additional guidance will be available in the future.

Section 3.4.1.3 of Chapter 10 states that the design will include features that support and facilitate a team approach.

3.4.2 Operating Procedures

Section 3.4.2 of Chapter 10 states that operating procedures will be developed as an integral part of the design. The design process for the operating procedures will include V&V. Where practical, the operating procedures will be presented on electronic displays. Where not practical and to supplement electronic displays, hard copies of operating procedures will be supplied. The staff's evaluation of the human factors aspects of this section is given in Appendix B to this chapter.

3.4.3 Selection of Automatic or Manual Control

Section 3.4.3 of Chapter 10 states that the criteria for choosing between automatic and manual control or monitoring operation will include consideration of

- operator workload
- operator capability
- experience
- operator vigilance

- complexity of hardware and software
- consequences of and potential for malfunctions
- regulatory requirements

The staff concludes that consideration of these items is appropriate.

3.4.4 Selection of Remote or Local Control

Section 3.4.4 of Chapter 10 states that the selection of local versus remote control and monitoring will include consideration of

- operator workload and access time
- operator capability and monitoring needs
- local environment
- local monitoring needs
- complexity, maintenance, and testing burden
- malfunctions of remote equipment
- experience

The staff concludes that consideration of these items is appropriate. The staff's evaluation of the human factors aspects of this section is provided in Appendix B of this chapter.

3.4.5 Operator Aids

Section 3.4.5 of Chapter 10 states that the design will specify features for operator assistance. These aids will be in the form of computer-aided displays and permanently posted information. The designer also will evaluate the incorporation of active systems that will predict the consequences of a potential action. In the DSER for Chapter 10, the staff concluded that EPRI should provide more guidance on the intentions and limitations of this requirement and identified this as an open issue. In Revision 5, EPRI revised Section 6.1.3.7.1, requiring that the complexity of safety-related software be minimized to the extent practical. The staff agrees that limiting unnecessary complexity for safety-related software is desirable and concludes that the requirements in Revision 4, Section 6.1.3.7.1 are acceptable. Therefore, the open issue is closed. The staff's evaluation of the human factors aspects of this section is provided in Appendix B to this chapter.

3.5 Availability and Reliability

Section 3.5 of Chapter 10 of the passive Requirements Document states that the designer will establish quantitative reliability and availability criteria for the component parts and subsystems. In its RAI dated April 10, 1990, the staff asked EPRI to explain how the quantitative reliability and availability criteria are to be determined. By letter dated July 23, 1990, EPRI responded that the estimates of core damage ($\leq$ 1x10E-5) and overall plant availability (87 percent over the life of the plant) are provided in Chapter 1 of the passive Requirements Document and it is up to the designer to allocate the reliability and availability to the various systems. The staff will evaluate the acceptability of the reliability and availability of the M-MIS during its review of an individual application for FDA/DC.

The passive Requirements Document does not describe how these estimates, especially software reliability, will be made with regard to the I&C design. It is not clear how these criteria will be established or used along with other defense-in-depth concepts to ensure adequate safety. In its RAI dated May 17, 1991, the staff asked EPRI to describe the process by which software reliability and availability will be quantitatively specified. In its response dated July 22, 1991, EPRI stated that an attempt to quantify software reliability is difficult. One approach to achieve highly reliable software is to use a rigorous design process that includes the use of experience, analysis, and testing at various stages to ensure that design errors will not significantly affect component and subsystem reliability. EPRI also noted that when designers aggregate component and subsystem analysis into a system-level model, the possibility of common-cause errors should be addressed. EPRI stated that the approach to measuring software reliability as a tool for planning and guiding software development is to consider both the software "product" and the software development "process" across the life cycle of the system. The staff would have EPRI consider the use of software metrics as a required design tool. In the DSER for Chapter 10, the staff concluded that EPRI should provide guidance regarding the establishment and use of these reliability and availability estimates and identified this as an open issue. By letter dated September 29, 1992, EPRI pointed out that Section 3.5 requires the use of quantitative reliability estimates for component parts and subsystems of the M-MIS. However, the staff concludes that this issue will be resolved during its review of an individual application for FDA/DC.

3.5.1 Effects of Postulated M-MIS Failures

Section 3.5.1 of Chapter 10 requires that the designer do a case-by-case evaluation of plant equipment and select the appropriate failure state on loss of power. It states that typically the protection systems' preferred failure state should be the safe condition. The control system will typically be designed to fail to the most stable state. The M-MIS will not change the state of the controlled components when power is restored and will initialize in the manual mode. In its RAI dated April 10, 1990, the staff asked EPRI to provide additional justification for the initialization requirements. The staff concludes that the protection systems should always, not typically, fail to the safe state. The determination of what the safe state is at all times may be difficult to establish in some cases. The rationale provided by EPRI that the fail-safe requirements in the past were implemented in such a way that the plant was in a safe mode only from a limited regulatory perspective does not appear to the staff to provide guidance to the designer on how to determine the safe state for any particular equipment. In its letter of July 23, 1990, EPRI stated that the designer will select, on a case-by-case basis, the appropriate failure state. Therefore, the staff will evaluate the failure modes selected by the designer during its review of an individual application for FDA/DC.

The passive Requirements Document requires that, to the extent practicable, the M-MIS and support equipment be single failure proof, including the capability for the online diagnostics and testing.

3.5.2 Top-Level Reliability Requirements

Section 3.5.2 of Chapter 10 requires the mean time between forced outages caused by failures of M-MIS equipment to be greater than 50 reactor operating years. This requirement includes any outages required to avoid violation of technical specifications. The mean time between failures of the M-MIS equipment that result in the reduction of plant availability will be greater than 5 years.

EPRI states that the mean time between failures that require corrective maintenance will be 14 days or more for the protection and safety, plant control, and plant information and monitoring systems. In its RAI dated April 10, 1990, the staff asked if this means that corrective maintenance of the protection system every 14 days is acceptable. By letter dated July 23, 1990, EPRI responded that the 14-day requirement is intended to include checks and potential adjustments or repairs that would be indicated by self-testing or self-diagnostics. The staff will evaluate the acceptability of the maintenance program during its review of an individual application for FDA/DC.

3.5.3 Design Requirements for Availability and Reliability

Section 3.5.3 of Chapter 10 provides guidance for segmenting M-MIS control and monitoring systems to protect against failures of M-MIS equipment. Segmentation is a major design requirement for the EPRI ALWR. If segmentation is not achieved, then EPRI's functional requirement to prevent propagation of unforeseen failures requires alternative design approaches that the designer must specifically identify and justify. EPRI states its intent to extend the segmentation and separation currently required for the safety and protection systems to the control and monitoring functions. The staff agrees with EPRI that this will improve the defense-in-depth aspects of the plant. Segmentation mitigates the reduction in the defense-in-depth from that in the existing analog designs that could result if central computers are used. The staff concludes that the guidance for achieving this requirement does not conflict with current regulatory requirements and is, therefore, acceptable.

Table 10.3-1 of Chapter 10 provides a list of control and monitoring functions that are required to be segmented. There are no specific NRC requirements for the segmentation of control systems; therefore, these are acceptable. However, the staff notes that these EPRI requirements prohibit an ICS, such as that used in reactors designed by Babcock and Wilcox.

Section 3.5.3.1 of Chapter 10 states that for each of the segments listed in Table 10.3-1, different sets of sensors, transmitters, and data communication paths will be used when possible. Cross-checking or calibration and compensation between functions will be allowed if the designer ensures that a complete failure in the instrumentation for one segment does not prevent the receiving control function of another segment from performing adequately.

In addition, EPRI states that for each segment, different processors and power supplies will be used whenever practicable. In its RAI dated April 10, 1990, the staff asked if the different processors will be diverse. By letter dated July 23, 1990, EPRI responded that the different processors do not need to be

diverse but simply a different physical piece of equipment. This is acceptable to the staff for the purposes of segmentation within a safety train. However, many of the other sections and the conclusions of this report describe the need for diversity in the design of the safety systems.

EPRI also requires that, whenever practicable, data processing and data communication electronic equipment be housed in separate enclosures. Equipment for segmented functions may share a common room. The staff notes that in addition to the requirements of this section, EPRI also has committed to physical-separation regulatory criteria, such as Appendix R to 10 CFR Part 50 and RG 1.75, "Physical Independence of Electrical Systems."

The different segments will be provided with separate power supplies and housed in separate enclosures to the maximum degree practical. In the DSER for Chapter 10, the staff noted that EPRI did not differentiate between "practical" and "maximum degree practical" and requested that EPRI clarify the distinction between these two terms. This was identified as an open issue. By letter dated September 29, 1992, EPRI explained that the terms "practical," "degree practical," and "maximum degree practical" are used in Section 3.5. Specifically, the term "maximum" was used in connection with the requirements of Section 3.5.3.1.4 to add special emphasis to the need to maintain the segmentation of multiplexers and similar data and control transmission equipment outside the MCR. This explanation clarifies the distinction between "practical" and "maximum degree practical." Therefore, this open issue is closed.

EPRI states that segmentation of the M-MIS is intended to mimic the plant system segmentation. As discussed above, the existing regulatory criteria still apply, and in addition, the segmentation guidelines in the passive Requirements Document will apply for safety systems.

Section 3.5.3.2 of Chapter 10 states that the designer will consider the potential for common-mode failures and their effects in determining the architecture and choosing hardware and software designs. The staff agrees that it is very important that potential hardware and software common-mode failures be addressed early in the design.

Section 3.5.3.3 of Chapter 10 states that the equipment will be designed for the environment that it is installed in and that passive methods and robust hardware are preferred. Equipment that is required to function following a loss of both onsite and offsite ac power will be designed to require only passive environmental control. In the rationale, EPRI notes that the reliability of the M-MIS equipment is sensitive to temperature and humidity; therefore, EPRI requires that a means be provided to alert plant operators to a loss of environmental control. The design of the M-MIS will be such that as long as the HVAC system is restored within 1 hour of failure, there will be no loss of function and no maintenance of M-MIS components will be required. EPRI has committed to meet the requirements of 10 CFR 50.63 (station blackout rule) in Chapter 1 of the passive Requirements Document. This may involve longer times without the HVAC system operable than those assumed in this section. In the DSER for Chapter 10, the staff concluded that the requirements to cope with loss of environmental control for individual cabinets versus that associated with loss of all ac power for 72 hours should be clarified. This was identified as an open issue and also was addressed in

Section 2.3 of Part I of the DSER for Chapter 10. In Revision 4, EPRI revised Sections 3.5.3.3 and 3.5.3.3.2 to require the M-MIS equipment, including the quality level of the components, to be designed to cope with a loss of environmental control for 1 hour. This revision clarifies the requirements for coping with loss of environmental control for individual cabinets versus that associated with loss of all ac power. Therefore, this open issue is closed.

3.5.4 Reliability and Maintainability Analysis

Section 3.5.4 of Chapter 10 states that the designer will perform analyses to predict the reliability of the M-MIS. These reliability analyses are to be consistent with MIL-HDBK-338, "Electronic Reliability Design Handbook," and MIL-HDBK-217E, "Reliability Prediction of Electronic Equipment." In the rationale, EPRI notes that these analyses are required to demonstrate that the M-MIS design meets the quantitative reliability and availability goals. In addition, Chapter 1 of the passive Requirements Document states that the M-MIS will meet the requirements of Section 5.3 of IEEE 603 regarding qualitative reliability goals for safety systems.

In its RAI dated April 10, 1990, the staff questioned the use of only these reliability analyses guidelines because they address the hardware issues but do not adequately address software reliability. The use of this guidance is acceptable to the staff for predicting hardware reliability but is not an acceptable method for predicting overall M-MIS reliability or availability. In its response dated July 23, 1990, EPRI stated that reliability data have been given in Appendix A to Chapter 1. The staff will evaluate the design-specific reliability analyses during its review of an individual application for FDA/DC.

Section 3.5.4.1 of Chapter 10 requires the designer to use the component reliability data in MIL-HDBK-217E or an equivalent source to quantitatively determine the mean time between failures. The resultant analyses will be independently verified. In the DSER for Chapter 10, the staff concluded that the use of this method alone was not acceptable because the components for the M-MIS are substantially software-based digital systems and identified this as an open issue. By letter dated September 29, 1992, EPRI pointed out that the reliability analysis and component reliability required in Section 3.5 are intended to be applied to hardware components only, not software. EPRI also pointed out that as discussed in the establishment and use of reliability and availability estimates, software quality is addressed through the software design process, and potential for software errors is considered in the analysis of potential common-mode failures as required by Section 3.5.3.2. The staff accepts EPRI's response and concludes that this issue will be resolved with each passive plant applicant during the staff's evaluation of an individual application for FDA/DC. Therefore, this open issue is closed.

EPRI states that the designer will perform reliability tests if adequate reliability data are not available in MIL-HDBK-217E or other equivalent sources. The general methodology for performing these tests is provided in MIL-STD-781, "Reliability Test Methods, Plans, and Environments of Engineering Development, Qualification, and Production." MIL-HDBK-338 will be used as a guideline for the statistical analysis. In the DSER for Chapter 10, the staff found this method acceptable for determining the reliability of the hardware

components, but did not consider it acceptable for determining overall M-MIS reliability. It required the M-MIS to meet the requirements of Section 5.3 of IEEE 603 for reliability goals. The staff concluded that this section should be clarified to reflect this position and identified this as an open issue. In its letter dated September 29, 1992, EPRI stated that, as discussed in establishment and use of reliability and availability estimates, the process of establishing overall M-MIS reliability and unavailability goals given in Chapter 1 proceeds through allocation of unavailability among systems and subsystems considering the specific reliability requirements in Chapter 10 Section 3.5. The process also uses classical methods for estimating reliability of M-MIS hardware. In addition, EPRI stated that Chapter 1 already requires that the M-MIS meet the requirements of IEEE 603 regarding reliability goal. The staff concludes that requiring IEEE 603 in Chapter 1 is acceptable; however, the staff will ensure that this issue has been properly addressed during its review of an individual application for FDA/DC. Therefore, the open issue is closed.

Section 3.5.4.2 of Chapter 10 specifies that the designer will perform analyses of all hardware and software to identify unplanned operational modes. This section refers to MIL-HDBK-338 as general guidance. In the rationale, EPRI defines a sneak circuit as a latent condition in a system that may result in an unexpected operation that is not due to equipment failure but rather to design oversight. The SCA is a method for reducing these design errors. EPRI states that, because of the complexity and size of many of the systems in the M-MIS, system testing to identify the sneak circuits is not an economical or logistically feasible alternative. It also notes that sneak circuits can appear in mature, thoroughly tested systems even after long periods of field use. In the DSER for Chapter 10, the staff concluded that because SCA could be a useful tool, an across-the-board elimination of system testing was unacceptable. This was identified as an open issue. In its letter dated September 29, 1992, EPRI explained that using SCA does not replace the system functional testing. EPRI also explained that the passive Requirements Document requires both system testing and SCA. This explanation clarifies the intent of SCA and is acceptable. Therefore, this open issue is closed.

Section 3.5.4.3 of Chapter 10 requires the designer to perform failure modes and effects analyses consistent with the guidance of MIL-HDBK-338 and MIL-STD-1629A, "Procedures for Performing a Failure Modes Effects and Criticality Analysis." This analysis will be used to demonstrate that the M-MIS meets the quantitative reliability and maintainability goals. The staff finds this acceptable, but notes that the requirements of IEEE 603 (which includes the requirements of IEEE 577 and IEEE 352) also apply.

Section 3.5.4.4 of Chapter 10 states that the designer will perform analyses of the M-MIS hardware to predict the amount of time that a system or component will be inoperative because of maintenance. The guideline to be used is MIL-HDBK-472, "Maintainability Prediction." The results will be used to determine if the goals of Section 3.7 of Chapter 10 of the passive Requirements Document have been met. The staff concludes that this is acceptable.

In its RAI dated May 17, 1991, the staff asked EPRI to address software maintenance requirements and responsibilities, and the effect of software maintenance on M-MIS availability. In its response dated July 22, 1991, EPRI stated that software maintenance is addressed in Sections 6.2.1 and 3.5 of

Chapter 10. In the DSER for Chapter 10, the staff concluded that software maintenance had not been adequately addressed and that EPRI should provide additional guidance. This was identified as an open issue. In its letter dated July 22, 1991, responding to the staff's May 17, 1991, RAI, EPRI referenced IEEE 1042 in Section 6.1.2.1.18 of Chapter 10. In addition, EPRI revised Section 6.1.7.3 in Revision 4 to reference IEEE 828 and 1042. The staff concludes that these standards provide adequate guidance on software maintenance. Therefore, this open issue is closed.

3.6 Testability Requirements

3.6.1 Continuous On-Line Testing

Section 3.6.1 of Chapter 10 of the passive Requirements Document states that the capability for continuous on-line testing of hardware integrity will be provided when practicable. These tests may include random access memory and read-only memory failure checks, arithmetic processing unit failure checks, data link buffer checks, and central processing unit reset of watchdog timers. In the DSER for Chapter 10, because it was concerned about necessary complexity, the staff concluded that the rationale needed to be clarified as to whether or not the tests listed were the minimum tests to meet technical specifications or other regulatory requirements. This was identified as an open issue. In Revision 5, EPRI added Section 6.1.3.7.1 to require that the complexity of safety-related software be minimized to the extent practical. The scope and extent of interfaces with other software functions such as those for operator aids, testing, and support must be designed to minimize the complexity of safety-related software logic and data structures. The staff agrees that the safety-related software complexity needs to be minimized to the extent practical and staff concludes that the addition of Section 6.1.3.7.1 resolves this concern. Therefore, this open issue is closed.

3.6.2 Periodic Testing

Section 3.6.2 of Chapter 10 states that capability for periodic testing will be provided and will meet the requirements of RGs 1.22, "Periodic Testing of Protection System Actuation Functions," and 1.118, "Periodic Testing of Electric Power and Protection Systems," and IEEE 338. The testing will be manually initiated but automatically performed. Automatic initiation of testing is allowed where it can be shown not to affect system function. The staff is not aware of any instances where these features proved to be undesirable; therefore, this requirement is acceptable. This type of test capability will also reduce the likelihood of transients initiated by human error.

3.6.3 Reliability of Testing Features

Section 3.6.3 of Chapter 10 states that the mean time between failures of the M-MIS continuous on-line self-test features and periodic functional test features will be greater than that of the equipment they are designed to test. The staff concludes that this requirement is acceptable because the test features will have a reliability equal to or greater than that of the equipment to be tested.

3.6.4 Reconfiguration After Failure Detection

Section 3.6.4 of Chapter 10 states that, if a failure is detected in the MIS, the system will be designed to automatically reconfigure, so that an additional single failure will not prevent system-level protection or safety action. The staff agrees with the concept of additional failure protection. However, it will review the details of the automatic reconfiguration setup during its review of an individual application for FDA/DC. In its RAI dated April 10, 1990, the staff asked if this requirement will allow a safety channel to be automatically bypassed and if operator notification will be via the non-safety annunciators. By letter dated July 23, 1990, EPRI stated that the configuration will not bypass the channel in the sense of preventing a trip; rather, it will set up the remainder of the system so that a protection trip will occur when needed, even if a second failure occurs. In addition, EPRI responded that although non-safety annunciation should be acceptable, annunciation of the bypass via the operator workstations would be via safety-grade equipment. The staff concludes that these requirements are acceptable because the single-failure criterion for the channels in service is maintained, and the annunciator equipment does not require additional isolation.

3.6.5 Failure Location Identification

Section 3.6.5 of Chapter 10 states that the test features will identify the location of the detected failure down to the lowest replaceable module. The staff agrees with this requirement.

3.6.6 Classification of Automatic Test Circuits

Section 3.6.6 of Chapter 10 states that the automatic testers will be classified as associated Class 1E as described in IEEE 384. IEEE 384-1981 sets forth criteria for the independence that can be achieved by physical separation and electrical isolation. This standard does not address software-based systems, and the staff has required that the software-based on-board diagnostics and automatic testers need to be qualified as Class 1E. Section 5.5.3 of IEEE 384 states that associated circuits need not be qualified for performance of function. The staff concludes that these automatic test circuits are being used to verify the correctness of the safety-grade hardware and software that cannot otherwise be verified and are, therefore, required to be the same classification. IEEE 384 requires the associated circuit to be the same quality as the Class 1E circuit and, for software, that would mean use of the same V&V procedures. Section 6.1.5.10 of Chapter 10 requires that all software and firmware used to meet the testability requirements in this section be of a level comparable to that of the base program software. The staff concludes that this meets the stated staff requirement for automatic test circuits.

3.6.7 System Reconfiguration for Testing

Section 3.6.7 of Chapter 10 states that built-in, automatically performed test features will be provided to eliminate physical reconfiguration of systems to accomplish required tests. The staff considers this to be a significant improvement over current operating plants and concludes that this requirement is acceptable.

3.6.8 Safety-Related System Testing

Section 3.6.8 of Chapter 10 states that the automatic testing features for the safety-related systems will be sufficient to meet the technical specification requirements for periodic surveillance as defined by RGs 1.22 and 1.118 and IEEE 338. The methods used to meet technical specification requirements will be established for specific designs. Therefore, the staff will evaluate the adequacy of the surveillance period of these testing features during its review of an individual application for FDA/DC.

3.6.9 Test Performance

Section 3.6.9 of Chapter 10 states that as much as possible, the test features of the M-MIS will be designed so that the tests can be performed at power. The staff agrees that test features should be incorporated into the design; therefore, this is acceptable.

3.6.10 Automatic Bypass

Section 3.6.10 of Chapter 10 states that when a test is initiated manually, the correct bypasses will be established automatically. RG 1.47, "Bypassed and Inoperable Status indication for Nuclear Power Plant Safety Systems," states (in the discussion section) that, generally, the plant's administrative procedures require that the operator's permission be obtained before any activity that would or could affect a safety-related system is initiated. The decision to grant such permission should be based on a knowledge of the operating status of the safety-related systems, the extent to which the activity will affect those systems, and whether the action is permissible within the provisions of the license. The RG lists the need for automatic indication of any bypass or inoperable status. It does not address automatic initiation of the bypass. Because this EPRI requirement does not violate the NRC criteria, the staff concludes that it is acceptable. However, the staff will evaluate the level of implementation during its review of an individual application for FDA/DC to ensure that the operators are aware of all automatic bypasses for each test initiated.

In its RAI dated April 10, 1990, the staff noted that in the rationale for this requirement, EPRI did not discuss the tradeoffs between operator error and potential increased software error. By letter dated July 23, 1990, EPRI stated that preliminary evaluations have led to the conclusion that the use of automatic testing improves plant reliability. The staff concludes that the requirements of Section 3.6.10 of Chapter 10 are acceptable.

3.6.11 Indicators for Test and Bypass Status

Section 3.6.11 of Chapter 10 states that local indication of pass or fail for the test and bypass status will be provided. Although not referenced here by EPRI, the staff concludes that the requirements of RG 1.47 will be applied to this indication.

3.6.12 Test Result Records

Section 3.6.12 of Chapter 10 states that a printer interface will be provided at the test cabinet to allow the operators to obtain a hard copy of the test results. The staff has no reason to believe that providing this feature as a convenience to the operators will have an adverse effect on plant operations or safety. Therefore, the staff concludes this is acceptable.

3.6.13 Removal of Automatic Bypass

Section 3.6.13 of Chapter 10 states that, upon completion of the testing, the bypasses will be automatically removed. Indication will be provided for the operators to verify bypass status. The staff concludes that this requirement is acceptable and will evaluate the acceptability of its implementation as part of its evaluation of automatic bypass initiation during its review of an individual application for FDA/DC.

3.6.14 Process Input Signals

Section 3.6.14 of Chapter 10 states that the safety system processors are required to remain in their normal execution paths; therefore, automatic testing will be performed using simulated process input signals. EPRI has not demonstrated to the staff that simulated inputs provide an accurate test for all real conditions. The staff will need to see specific implementation information before it can find this method acceptable. It will evaluate the acceptability of this information as part of its evaluation of automatic bypass initiation during its review of an individual application for FDA/DC.

3.6.15 Testing at Initialization of Processors

Section 3.6.15 of Chapter 10 states that comprehensive self-diagnostic routines will be performed upon initialization for all processors. In its RAI dated April 10, 1990, the staff asked EPRI to evaluate the time delay impact this may have relative to the requirements of Section 3.5.1.1 of Chapter 10 of the passive Requirements Document. By letter dated July 23, 1990, EPRI responded that any additional time required to get the system on line and operational will be considered in the mean time-to-repair requirements. The staff concludes that this is acceptable.

3.7 Maintainability

Section 3.7 of Chapter 10 of the passive Requirements Document states that the M-MIS will be designed for simplification of maintenance over the lifetime of the plant. The staff agrees with the increased emphasis on maintenance concerns during the design phase.

3.7.1 Maintenance Burden

Section 3.7.1 of Chapter 10 states that the effect of maintenance activities on the operator and maintenance personnel will be quantified.

3.7.2 Replacement of Equipment

Section 3.7.2 of Chapter 10 states that service life and replacement needs will be considered in the design.

3.7.3 Modular Replacement

Section 3.7.3 of Chapter 10 states that the M-MIS is to be designed so that normal repair activities will consist of simple modular replacement in the field with module repair in the shop.

3.7.4 Time To Detect and Repair a Failure

Section 3.7.4 of Chapter 10 states that the time to detect and replace a faulty module will be less than 4 hours average with a maximum of 8 hours for any single module. This requirement will apply to equipment with self-test capability. Any specific technical specifications still will have to be met. The staff notes that this requirement applies only for those failures that can be repaired by module replacement. In its RAI dated April 10, 1990, the staff asked how these requirements pertain to reliability and how the detection of software errors and the correction of errors will be considered. By letter dated July 23, 1990, EPRI responded that this section does not address correction of software errors. EPRI considers those to be design errors that are addressed through the V&V process. The staff believes that module software concerns and out-of-service time can be addressed in technical specification requirements for specific designs; therefore, the staff concludes that this requirement is acceptable. It will address the technical specification issue during its review of an individual application for FDA/DC.

3.7.5 On-Line Calibration

Section 3.7.5 of Chapter 10 states that any module that requires calibration more often than during each scheduled refueling outage will be capable of being calibrated on line while maintaining control, monitoring, and system performance requirements.

3.7.6 On-Line Maintenance and Repair

Section 3.7.6 of Chapter 10 states that expected maintenance or repair will not prevent any plant safety or protection system from fulfilling its requirements. On-line repair is preferred. The staff will evaluate bypass and test lockouts for the channel that is not in repair during its review of an individual application for FDA/DC.

3.7.7 Maintenance Human Factors

Section 3.7.7 of Chapter 10 states that human factors with regard to maintenance will be considered in the M-MIS design by utilizing EPRI NP-4350, "Human Engineering Design Guidelines for Maintainability." The staff's evaluation of the human factors issues is given in Appendix B to this chapter.

EPRI states that the designer will identify the maintenance tasks as part of the design and will evaluate these tasks to ensure that they are simple and well understood. Equipment will be labeled in accordance with the general

guidance of EPRI NP-6209, "Effective Plant Labeling and Coding." In general, the equipment will be designed to facilitate maintenance activities. Access for maintenance personnel will be considered during the design process.

Section 3.7.7 of Chapter 10 states that maintenance activities will be designed so that they do not interrupt the operator at manned control stations. The M-MIS equipment will be designed to facilitate testing and repairs without requiring the operator to assist. The operators will be provided with indication that testing or repairs are being performed. Controls and displays used only by maintenance personnel will not be located on panel fronts unless they are covered and do not crowd the operator's controls.

3.8 Constructibility

Section 3.8 of Chapter 10 of the passive Requirements Document specifies that features will be included in the M-MIS design to reduce construction time and effort.

3.8.1 Use of Proven Techniques

Section 3.8.1 of Chapter 10 states that proven manufacturing, assembly, and installation techniques will be used for the fabrication and installation of M-MIS equipment.

3.8.2 Minimization of Field Operations

Section 3.8.2.1 of Chapter 10 states that the M-MIS is to be designed to facilitate installation. Each module will allow installation and functional testing before integration of the complete system. Shop fabrication of modules will be used, where possible. The amount of field wiring will be minimized.

Section 3.8.2.3 of Chapter 10 states that spare conductors will be provided in wire harnesses and cables where practicable. The staff considers the inclusion of spares for possible future modifications to be good design practice. EPRI states that damaged multiconductor cable will normally require complete replacement. The staff concludes that this is acceptable.

3.8.3 Standardized Designs for Construction

Section 3.8.3 of Chapter 10 states that the equipment used in construction such as connectors and labels will be standardized as much as possible.

3.8.4 Schedule for Construction

Section 3.8.4 of Chapter 10 states that the construction, fabrication, and installation of the M-MIS will support the overall plant schedule construction.

3.9 Design Flexibility

Section 3.9 of Chapter 10 of the passive Requirements Document states that the design will provide flexibility for changes or replacement by including spare capacities and modular design.

In its RAI dated May 17, 1991, the staff asked EPRI to provide guidelines for the amount of excess capacity deemed sufficient. In its response dated July 22, 1991, EPRI stated that it will be the M-MIS designer's responsibility to select the specific amount of spare capacity needed for each aspect of the M-MIS. (The staff notes that this is inconsistent with Section 5.2.3 of Chapter 10, which provides specific excess capacity values.) However, since NRC criteria do not address spare capacity requirements, the staff finds this acceptable.

3.10 Conclusions

The staff concludes that the requirements in Section 3 of Chapter 10 do not conflict with current regulatory requirements and, therefore, are acceptable. However, the requirements by themselves do not provide sufficient information to make a determination that a specific design application will be acceptable. Therefore, applicants referencing the passive Requirements Document will be required to demonstrate compliance with the guidance in the SRP, or provide justification for alternative means of implementing the associated regulatory requirements.

4 CONTROL STATION REQUIREMENTS

4.1 Control Station Design Process Requirements

Section 4.1 of Chapter 10 of the passive Requirements Document indicates that the design of the M-MIS will be consistent throughout the plant. The staff's evaluation of the human factors aspects of this section is given in Appendix B to this chapter.

4.1.1 Utilization of Functions and Tasks

Section 4.1.1 of Chapter 10 states that the control station will be designed on the basis of the overall identification of functions and tasks listed in Section 3.1.3.3 of Chapter 10.

4.1.2 Control Station Conceptual Design

Section 4.1.2 of Chapter 10 states that, following the initial definition of tasks, a conceptual design for each control station will be prepared. This conceptual design will include a layout drawing of the station, identification of the major control and display characteristics, a specific listing of tasks in sequence, requirements for information and control, and preliminary procedures for control station operation.

4.1.3 Review of Conceptual Designs

Section 4.1.3 of Chapter 10 states that the interdisciplinary design group will review each control station conceptual design. The control station review team will include human factors and operation specialists in addition to engineers in the traditional disciplines.

A static mockup of each control station will be used in the review process so that changes can be made easily and evaluated. The review process will provide for the use of active simulation of the control stations to provide time input and operational context review. The staff concludes that this is acceptable. The staff's evaluation of the human factors aspects of this section is given in Appendix B to this chapter.

4.1.4 Iteration of Functions, Tasks, and Designs

Section 4.1.4 of Chapter 10 states that the design process will provide for feedback from the control station design to the original overall identification of functions and tasks to correct any unsatisfactory control station designs. The staff concludes that this is good engineering practice and is, therefore, acceptable. The staff's evaluation of the human factors aspects of this section is given in Appendix B to this chapter.

4.1.5 Definition of Design Practices

Section 4.1.5 of Chapter 10 states that the design will follow the guidance on human factors practice, such as that provided in EPRI NP-3659, "Human Factors Guide for Nuclear Power Plant Control Room Development," and EPRI NP-6209, "Effective Plant Labeling and Coding." These guidelines address the subjects

that are to be considered during the design process but leave the details and selections to the designer. When the designer selects new technology for which published human factors guidance is limited, EPRI states that the designer will develop the necessary design practices based on the best available information. These design practices will be verified by experimentation and simulation. This requirement is vague and appears to contradict the EPRI requirement to use proven technology. As do many of the requirements given in the passive Requirements Document, this one will admit many designs or technologies. However, the staff concludes that this requirement is acceptable because it does not conflict with current regulatory requirements. The staff's evaluation of the human factors aspects of this section is given in Appendix B to this chapter.

4.1.6 Documentation of Final Designs

Section 4.1.6 of Chapter 10 specifies that the final design of the control stations will be documented in detail. The staff concludes that this is a good design practice.

The design configuration will be presented in a format useful to the operators. The design documentation for each control station will define the functions and tasks for that station. EPRI states that the design documentation will also define the common design practices used for all stations and will document any differences for a specific station. It will also include the control station operating procedures. The staff's evaluation of the human factors aspects of generic operating procedures is given in Appendix B to this chapter.

4.2 Operating Crew

Section 4.2.1 of Chapter 10 of the passive Requirements Document states that the M-MIS design will accommodate staffing for normal operation. The designer will specify the responsibilities of the operating crew members assumed in the design for use by the utility in staffing and training. The staff's evaluation of the human factors aspects of this section is given in Appendix B to this chapter.

In its RAI dated April 10, 1990, the staff asked EPRI to address the staffing requirements for maintenance personnel because the mean time-to-repair estimates are relatively short for complex equipment. By letter dated July 23, 1990, EPRI responded that the specific maintenance personnel requirements will not be defined until experience with the new technology is gained. The staff concludes that maintenance staff requirements should be developed during the testing and validation of newer designs. Since the technical specifications will address the time that a system can be out of service rather than a required maintenance staffing level, this requirement is acceptable.

Section 4.2.2 of Chapter 10 provides a list of assumptions regarding operator staffing during plant transients and emergencies that are to be used in performing the analyses required by EPRI in Section 3.1.3.3.2 of Chapter 10. The staff concludes that these assumptions are consistent with current regulatory requirements for licensed operators and are, therefore, acceptable.

Section 2.3.3 of Chapter 1 requires that the time for an operator to react in an emergency will not be less than 30 minutes. However, the design will not prevent the operator from taking action before 30 minutes. In addition, no credit for operator action will be necessary to meet regulatory limits for at least 72 hours following the initiating event. The staff concludes that these requirements do not conflict with regulatory requirements, and are, therefore, acceptable.

Section 4.2.3 of Chapter 10 specifies that the MCR will support a maximum of eight crew members and provisions will be made for three active observers, one of whom is from the NRC, one from plant owners management, and one to handle communications during emergencies. Current agency procedures (NUREG-0845 and Information Notice (IN) 86-18, "NRC On-Scene Response During a Major Emergency") indicate that one NRC observer will initially report to the control room in an emergency (or TSC, if activated), and that two NRC observers (with potential for a third person) will be located in the control room as part of an expanded site team. EPRI states that a more effective location for the second NRC observer may be the TSC because of the requirements in Section 4.9.4 that TSC personnel have access to all the monitoring data available to the control room operators and that they be able to visually monitor activities in the MCR (either by direct viewing or closed-circuit TV). Additionally, in its response to the NRC on January 28, 1992, EPRI stated that the specific work areas for the observers in the MCR will allow for temporary occupancy by two persons when the personnel are being relieved. The staff finds that the EPRI concept for accommodating an expanded NRC site team in the event of an emergency is acceptable.

However, recent operating experience (IN 91-77, "Shift Staffing at Nuclear Power Plants") has indicated that it may be necessary during emergencies for the licensee to expand its shift staffing beyond that indicated by EPRI in Section 4.2.3 of Chapter 10. EPRI should retain the flexibility to accommodate additional staffing in the MCR in the event of an emergency. The staff will review an individual application for FDA/DC to ensure that sufficient space has been provided in the MCR to accommodate persons needed to adequately respond to all events.

Section 4.2.4 of Chapter 10 states that the M-MIS will be designed so that a single operator can perform all normal functions, including taking the plant from hot standby to full power and back again. In the rationale, EPRI states that this requirement is not meant to require a single operator for plant startup. EPRI expects that if operations can be performed under normal, routine circumstances by a single individual, then the additional personnel who will actually be available will assuredly be free to handle upsets and emergencies. The minimum number of licensed operators required to be on shift and on site is specified in 10 CFR 50.54(m).

Section 4.2.5 of Chapter 10 states that the M-MIS will be designed so that the normal shift crew can start up and shut down the plant. Operations at local control stations are still expected for these operations.

Section 4.2.6 of Chapter 10 states that the design will be based, whenever practical, on the operator skill levels at existing plants or the designer will specify any differences in these levels.

4.3 Alarms

4.3.1 General Alarm System Requirements

Section 4.3.1 of Chapter 10 of the passive Requirements Document states that the M-MIS will include a main process alarm system and local alarm systems. The alarm systems will alert the operators to off-normal conditions, guide them to the appropriate response, and assist them in determining the state of the plant. Emphasis will be placed on minimizing distraction and unnecessary workload. EPRI states that the design will follow the guidance of EPRI NP-3448, "A Procedure for Reviewing and Improving Power Plant Alarm Systems," and EPRI NP-3659, "Human Factors Guide for Nuclear Power Plant Control Room Development."

The function and task analysis will specifically address alarm information. The alarm system will be testable to verify its operating status. The effectiveness of the alarm system will be verified with a real-time, dynamic simulator. EPRI states that the designer will define the measures used to judge the effectiveness of the alarm system early in the design process. Recent operational experience has shown a vulnerability in the power supplies for the alarm system. Therefore, in the DSER for Chapter 10, the staff concluded that EPRI should reevaluate the design basis and reliability for alarm system power supplies, and identified this as an open issue. In its letter dated September 29, 1992, EPRI stated that Section 6.2.8 provides requirements that are applicable to all M-MIS power supplies and cover a wide variety of attributes. In addition, Section 2.2.10 provides alarm requirements that are consistent with those stated in the policy issue on control room annunciator reliability in SECY-93-087. The staff notes that Section 4.3 primarily discusses the alarm system functionality, while reliability requirements are discussed in other sections of the passive Requirements Document. The staff concludes that EPRI has adequately addressed the vulnerability of power supplies for alarm systems. Therefore, this open issue is closed.

4.3.2 Selection of Alarm Conditions

Section 4.3.2 of Chapter 10 indicates that a consistent approach will be used in selecting plant conditions that are to be alarmed. This is a good engineering practice and is, therefore, acceptable.

The alarms will be presented in a "dark board at power" format; that is no alarms should be present when the plant is operating normally at full power, with all systems in their normal configuration. The passive Requirements Document references EPRI NP-3448 for guidance. In its RAI dated April 10, 1990, the staff asked EPRI to describe the method that would be used to verify operability of the dark-board alarms. In its July 23, 1990, letter, EPRI responded that the method would be similar to the current method, that is, primarily periodic testing. This is consistent with current regulatory practice and is, therefore, acceptable.

EPRI states that the designer will outline a response procedure for each alarm condition.

The operator will be provided the capability to establish temporary alarms and setpoints for specific conditions. In its RAI dated April 10, 1990, the staff

asked EPRI to provide additional justification for this requirement. Although the staff encourages operator input during the design, operator manipulation of software, alarms, and setpoints during plant operation needs to be evaluated in detail. By letter dated July 23, 1990, EPRI stated that this requirement is limited to providing the capability for additional operator-defined alarms to monitor special, temporary situations without affecting the required alarms. This requirement does not conflict with current regulatory requirements and is, therefore, acceptable.

EPRI states that the designer will document the basis for each alarm selection and will follow the guidance in EPRI NP-3448.

4.3.3 Alarm Processing

Section 4.3.3 of Chapter 10 states that the alarm system will be designed to minimize the potential for nuisance alarms and to allow time filtering or delays and conditioning of the input signals. EPRI states that the individual alarms will be evaluated to examine the potential for nuisance alarming. The following will be considered in the evaluation:

- all modes of operation
- maintenance of the associated system or equipment
- possible momentary alarms
- system dynamic response to plant transients or upsets
- alarm input noise
- unusual system lineups
- other conditions

On the basis of the evaluation to the criteria listed above, EPRI states that the designer will select the alarm setpoint, delays, conditioning logic, and other features. The designer will ensure that the elimination of nuisance alarms will not prevent the signal from an alarm when it is actually needed. This is acceptable.

EPRI indicates that an alarm that has more than one input will have the capability to actuate again if an alarm condition occurs after a first alarm has signaled and been acknowledged. This is a desirable feature and is, therefore, acceptable.

The passive Requirements Document states that the alarm system will be designed to minimize the number and rate of alarms during plant upsets. The primary method to be used will be the use of component- and system-based logic to make the alarms less likely to occur unnecessarily. Alarm suppression schemes are allowed if justified by the designer. This requirement is acceptable. However, the staff will evaluate the methods to be used and the details of the suppression schemes during its review of an individual application for FDA/DC.

EPRI states that the alarm system will include features or capabilities that are appropriate for dealing with alarms that are out of service.

4.3.4 Alarm Presentation

Section 4.3.4 of Chapter 10 provides requirements to ensure that alarm information will be presented so that both individual and crew needs are supported. The alarms will be integrated with the controls and displays for operator ease of use.

EPRI states that the alarms that require a short response time by the operator will be spatially dedicated, continuous, and located with the controls and displays for the function that is alarmed. Both the normal and alarm state will be presented. The alarms will be grouped by system or function. EPRI indicates that the designer should use the guidance provided in EPRI NP-5693 and NP-3448. Alarms will be prioritized. The priorities will be established on the basis of criteria that will be part of the design practices. In the DSER for Chapter 10, the staff concluded that this requirement did not conflict with current regulatory requirements and was, therefore, acceptable. However, since it did not provide any criteria for assignment of priorities, EPRI should provide such guidance. This was identified as an open issue. In its letter dated September 29, 1992, EPRI stated that prioritization of alarms will be based on relative importance or urgency and the time within which the operator must take action and that it considers the requirements in Section 4.3.4 of Chapter 10 adequate to define the features that should be provided by the alarm prioritization scheme. EPRI also stated that defining prioritization of alarms and getting a well-balanced and integrated alarm system should be a result of the design process. The staff agrees with EPRI's response. Therefore, this open issue is closed.

EPRI states that the number of alarms assigned the highest priority will be limited. In the DSER for Chapter 10, the staff concluded that this requirement was acceptable, but that there was no specific guidance as to the maximum limit that would be considered acceptable. It stated that EPRI should provide such guidance and identified this as an open issue. In its letter dated September 29, 1992, EPRI pointed out that the quantitative limit on the number of high-priority alarms is a function of the specific scheme used to present the alarms and that these characteristics will be defined in detail as part of the design process. The staff concludes that EPRI has provided adequate guidance on the minimum number of alarms. Therefore, this open issue is closed.

The display capacity used for the alarms will not be exceeded during accident situations, and paging of the display will not be required to view all of the highest priority alarms. The prioritization will be evaluated using real-time simulation.

EPRI states that the alarm system controls will be located so that the operator can respond adequately and so that they can be read by the operator from the control station being used to respond to the alarm. The alarm system will have different tone capability so that the operator will be able to tell from the tone and the direction of the sound the priority and general system involved. The alarms will be treated as an integral part of the design.

Section 4.3.4.4 of Chapter 10 states that each alarm will be tagged to a resolution of 2 seconds or less. Points that are specifically designated as a sequence-of-events alarms will have a resolution of 4 milliseconds or less unless the designer demonstrates a longer time resolution is acceptable.

At least 4 hours of recorded alarm sequence will be provided. The time sequence of alarms will be kept as a permanent record of plant operation. In the DSER for Chapter 10, the staff concluded that this requirement was acceptable as long as a method is provided to make copies of the alarm sequence so that a continuous alarm sequence will be available for events exceeding 4 hours. It stated that the passive Requirements Document should be revised to add this requirement and identified this as an open issue. In its letter dated September 29, 1992, EPRI stated that the existing provisions in Chapter 10 ensure that a complete record of alarms (greater than 4 hours) will be provided. The purpose of this alarm record is to help the operators understand events leading up to and causing an upset. The time sequence of all alarms for historical purposes and post-trip review is covered by Section 4.3.4.8 of Chapter 10. Further, requirements on the storage of the historical data are provided in Section 6.1.3.20. The staff concludes that EPRI has adequately addressed the issue of alarm records. Therefore, this open issue is closed.

4.4 Displays

Section 4.4 of Chapter 10 of the passive Requirements Document specifies that the displays will be designed using the guidance of EPRI NP-3659 and NP-3701, "Computer-Generated Display Guidelines" (Volumes 1 and 2). In its RAI dated April 10, 1990, the staff asked EPRI to address how the display requirements in RG 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environ Conditions During and Following an Accident," will be implemented. By letter dated July 23, 1990, EPRI responded that the requirements will be met with the normal qualified instrumentation. Some items will be on the overview panel in addition to the workstations. Section 4.9.1.8 of Chapter 10 requires postaccident monitoring to be accomplished with normal plant instrumentation. This is acceptable.

Section 4.4.1 of Chapter 10 indicates that the number of different types of displays will be minimized.

Section 4.4.2 of Chapter 10 states that position or status information will be provided directly rather than through the demand signal. This is a generally desirable feature and is, therefore, acceptable.

Section 4.4.3 of Chapter 10 indicates that displays will clearly indicate loss of functionality. Loss of power to a meter will not result in failing to midscale, which could be interpreted by the operator as a valid signal.

Section 4.4.4 of Chapter 10 originally specified that position indication is to be provided for all valves at the location where they are controlled. Yet this same section later stated: "Position indication for manually operated valves may be provided if required by the analysis of functions and tasks." In the DSER for Chapter 10, the staff concluded that this could be misinterpreted to allow some manual valves to have no position indication if not required by functions and task analysis. It stated that EPRI should clarify

this section to indicate that valve position indication will be provided for all valves, including manual valves, at any location where the valves can be controlled, and, in addition, at any location where functions and task analysis suggest it is necessary to provide assurance of proper system alignment. This was identified as an open issue. In Revision 3, EPRI changed the third sentence in Section 4.4.4 of Chapter 10 from "Position indication for manually operated valves . . . ." to "Remote position indication for manually operated valves . . . ." EPRI also explained that Section 12.3.2.3.3.2 of Chapter 1 specifically requires each manual and power-operated valve to have positive, mechanical indication of its overall position (i.e., open or closed), which can be determined by direct observation at the valve without instruments or power. This includes requirements for the position indication at the valve for partially open valves. The staff concludes that this revision clarifies the requirements for valve position indication. Therefore, this open issue is closed. See Section 6.2.7 and Appendix B of this chapter for further discussion of requirements for valve indication.

Section 4.4.5 of Chapter 10 states that indication of current draw will be provided for any major motor that can be started from the control station. The indication will identify the normal starting and running current draw. This does not apply to short-duration motors.

Section 4.4.6 of Chapter 10 states that indicator lights that use a single light element such as incandescent bulbs and light-emitting diodes will not be used.

Section 4.4.7 of Chapter 10 states that conventional paper-and-ink strip chart recorders will not be used. The data-recording function of the recorders will be provided by the plant data system.

Section 4.4.8 of Chapter 10 states that the designer will evaluate the need for coordinating the tasks of multiple workstations when the stations are located in the same room. Displays will be included in the design process. The MCR will include a display that will provide the values of key operating parameters, including power level, pressure, temperature, and flow rate indication, and the operational status of essential components that are controlled or monitored from the control room.

EPRI states that the design will provide for the display of the operational status of essential components controlled and monitored from the MCR. The display will also display high-level derived quantities and provide for spatially dedicated display of certain key alarms.

EPRI states that active simulation will be used to validate display design practices and presentation guidelines. The display will provide information needed to support use of any manual system-level actuation controls. EPRI addresses the need for overview display information when the compact workstations are not available. In its RAI dated May 17, 1991, the staff asked EPRI to define a set of conditions during which all compact workstations could be inoperable without losing the overview display. In its response dated July 22, 1991, EPRI stated that without a detailed design for the workstations and the system architecture to support them, the conditions that could lead to

all workstations being inoperable cannot be defined. The staff will evaluate this issue by defense-in-depth and common-mode-failure analysis during its review of an individual application for FDA/DC.

Section 4.4.9 of Chapter 10 states that displays of plant parameters and component status on workstations will be produced by electronic techniques such as cathode ray tubes. This is consistent with the EPRI policy statement for the M-MIS.

Section 4.4.10 of Chapter 10 specifies that closed-circuit television (CCTV) will be used for areas that are not easily accessible to the operators because of the need to keep radiation exposure of personnel as low as is reasonably achievable and other considerations. This is acceptable.

The staff's evaluation of the human factors aspects of this section is given in Appendix B to this chapter.

4.5 Controls

Section 4.5 of Chapter 10 of the passive Requirements Document requires that the designer develop specifications for the M-MIS that are consistent with the guidance of EPRI NP-3659. As noted in the rationale, this guidance does not prescribe the design details but rather identifies the attributes that have to be considered by the designer.

Section 4.5.1 of Chapter 10 states that the M-MIS design will minimize the number of different types of controls as much as possible.

EPRI requires that each control identify the power source of the controlled device. However, Section 4.5.2 of Chapter 10 does not give any specific method for accomplishing this task.

Section 4.5.3 of Chapter 10 specifies that the normal control position will be identified consistently.

Section 4.5.4 of Chapter 10 states that measures other than key-locked controls will be used to prevent inadvertent actuation of controls, if practical, and requires that key-locked controls not be used for controls that may need to be actuated on a timely basis in response to an emergency.

It is staff's position that key-locked controls would be appropriate for some security system controls. (The requirements in Chapter 10 relative to ensuring the functionality of the man-machine interface are stated as also applicable to the plant security systems.) In its letter dated July 22, 1991, EPRI pointed out that Section 4.5.4 discourages but does not preclude the use of key-locked controls. The rationale of Section 4.5.4 allows a key-locked control if it is the only practical method to ensure that a control is not operated inadvertently or by unauthorized personnel. EPRI also noted that the results of the sabotage vulnerability analysis required by Section 5.2.2.1 of Chapter 9 would also have to be considered by the designer of the M-MIS for the security system in the selection of the method to prevent inadvertent or unauthorized operation for a particular control. This is an acceptable means of resolving this issue and will be evaluated during the staff's review of an individual application for FDA/DC.

Section 4.5.5 of Chapter 10 states that controls of plant components and systems on the MCR workstations will normally be electronic or "soft" controls. This does not apply when there are specific requirements to the contrary.

Section 4.5.6 of Chapter 10 states that automatically actuated safety functions will be provided with manual system-level actuation controls. These manual controls must be independent of the operability of workstations, physically separated from other controls, hardwired to the lowest practical level, accessible to operators, protected against inadvertent actuation, simplistic, and located so that the operator has feedback after the actuation. The staff concludes that the requirements of Section 4.5 of Chapter 10 of the passive Requirements Document specify desirable attributes for controls and are, therefore, acceptable. However, the staff would require a vendor to analyze the design to ensure that it provides adequate defense-in-depth and diversity. The staff will address this issue during its review of an individual application for FDA/DC.

The staff's evaluation of the human factors aspects of this section is given in Appendix B to this chapter.

4.6 Voice Communication Systems

Section 4.6.1 of Chapter 10 of the passive Requirements Document states that the scope of Section 4.6 includes both in-plant communications for operations and maintenance and communications outside the plant, such as communication with a load dispatcher as well as with agencies for which communications are needed to support emergency operations. Requirements for specific security system communications are included in Chapter 9. As clarified in EPRI's letter dated October 12, 1990, the staff agrees that the general communication requirements of Chapter 10 are also applicable to security communication systems.

Section 4.6.2 of Chapter 10 states that the primary, dedicated means of communication between operators during normal or emergency operation will be by portable, wireless communication equipment supported by appropriate base stations, antennas, amplifiers, and/or repeaters. A plantwide paging system and inplant telephone system will be included. Offsite communications will be accomplished primarily with dedicated phone links. The staff notes that extensive wireless communication will increase the emphasis on consideration of electromagnetic interference (EMI) in the design of the M-MIS, but the use of this type of communication system is not specifically excluded in NRC regulations, may be necessary to meet the requirements of 10 CFR Part 73, and is, therefore, acceptable.

This section, in addition to identifying areas for the designer to evaluate, specifies that there will be a frequency allocation plan to ensure that there is no interference between communication systems and to ensure compatibility with the EMI/radiofrequency interference (RFI) measures taken by the electronics and computer designers. In the DSER for Chapter 10, the staff stated that this was a good engineering goal and was, therefore, acceptable. However, no specific guidance on methods to achieve this goal was provided. The staff stated that EPRI should provide such guidance and identified this as an open issue. In its letter dated September 29, 1992, EPRI stated that the

allocation of frequencies requires detailed technical information that is highly dependent on the specific plant and equipment design. EPRI also stated that it is not practical to provide specific guidance or requirements on the frequency allocation plan itself at this time. Because EPRI is unable to provide detailed guidance on the frequency allocation plan, the staff will address this issue during its review of an individual application for FDA/DC. Therefore, this open issue is closed.

Section 4.6.2.2 of Chapter 10 requires the designer of the communication system to include an analysis of the specific communication needs and the specific design requirements in the design-basis documentation. In response to a staff comment, EPRI modified this section to include in this analysis the consideration of specific communication tasks that must be performed by security personnel. Additional communication needs for security operations are addressed in Chapter 9, and the interface with Chapter 10 is noted therein. Requirements for security communications are also specified in 10 CFR 73.55(f).

Section 4.6.3 of Chapter 10 indicates that the method preferred by EPRI for dedicated wireless communications is a point-to-point method similar to cellular telephones where each receiver can be specifically addressed. The capability to have open multiparty communication will exist. Adequate equipment will be provided to ensure clear communications.

EPRI states that potential high-noise areas will be considered in the communication system design. The plant designer will provide adequate means to alert personnel in high-noise environments that they are to use the communication systems.

Portable communications equipment will be integrated with protective equipment, including diving suits and respirators, so that personnel wearing this protective equipment will be able to communicate effectively. EPRI states that the design will apply the guidance developed by EPRI on voice communications systems compatible with respiratory protection (EPRI NP-6559, "Voice Communication Systems Compatible with Respiratory Protection"). This is good design practice and is, therefore, acceptable.

Section 4.6.3 of Chapter 10 states that an ample (unspecified) number of communication channels will be provided.

This section stipulates that there will be no interference between the communications systems and M-MIS equipment. The communications designers are required to define the worst-case emissions from the communications equipment, including the type, magnitude, frequency content, and locations. All potential uses of the communications gear will be considered. EPRI states that adequate protection will be provided for the M-MIS equipment. Maintenance activities will be considered. The communications system will also be protected from the M-MIS equipment to the extent that clear communications will be maintained. The staff agrees with this goal and concludes that it is acceptable.

However, because of the potential effects of EMI/RFI on the M-MIS equipment, the staff stated in the DSER for Chapter 10 that EPRI should provide design guidance regarding implementation of this goal and identified this as an open

issue. In its letter dated September 29, 1992, EPRI explained that Section 4.6.3.6 requires that no interference exist between communication systems and the M-MIS equipment. This section provides the needed interference guidance; therefore, this open issue is closed.

Section 4.6 requires an extensive, fairly powerful, portable, wireless communications capability. This capability will require extensive EMI/RFI testing of all safety electronics and probably should require testing of all control electronics for susceptibility. Existing, otherwise-proven equipment has shown definite susceptibility to wireless systems significantly less powerful than this proposed system. EPRI states in Section 4.6.3.6 that testing will be performed to verify that adequate measures have been taken to control interference. This is acceptable.

Section 4.6.3 also states a preference for telephone-type dial-up wireless equipment, with a requirement for open channel or "party line" communications for situations that require this. The staff considers dial-up telephone-type portable radios to be unsuitable as the primary equipment for security armed response personnel. In its letter dated July 22, 1991, EPRI stated that it is evident that the specific security communication requirements of Section 5.2.1.1 of Chapter 9 will require some special communication equipment dedicated to security-related uses. As noted above, EPRI considers that the analysis of the specific communication needs of the security system required of the communication system designer will ensure appropriate communication equipment for security personnel. The staff considers this to be an acceptable means of resolving this issue and will evaluate this matter during its review of an individual application for FDA/DC.

EPRI states that the communications system will be designed to be highly reliable and resistant to failures and that it will be powered from reliable, backed-up sources of onsite power.

Section 4.6.4 of Chapter 10 states that the communications station in the MCR will be designed so that the operator can use more than one station easily and the communications equipment will not impede use of the monitoring or control equipment.

Different tones or lights will be used to alert the operator to specific communications equipment. The communications signals will be easily distinguishable from the other signals and alarms at the control station. The communications equipment will be identified so that it can be easily located.

Adequate voice communication will be possible between control room operators and technicians working in the control room. Because Section 4.7.6 of Chapter 10 specifies an ambient noise level of less than 60 dB(A) and the control room is designed to be relatively small, this requirement will be achievable and it is acceptable.

The staff's evaluation of the human factors aspects of this section is given in Appendix B to this chapter.

4.7 Arrangement, Environment, and Equipment

Section 4.7 of Chapter 10 of the passive Requirements Document states that the arrangement of the M-MIS will be designed in accordance with the guidance in EPRI NP-3659, "Human Factors Guide for Nuclear Power Plant Control Room Development." This guidance addresses those items that should be considered rather than providing specific design guidance.

EPRI states that the control stations will be arranged and have an environment and equipment suitable for the functions and tasks assigned it. This is a good design practice and is, therefore, acceptable.

Section 4.7 of Chapter 10 includes provisions that the control stations will also have features that will enable the operators to complete the reporting, logging, and related activities that do not directly involve equipment control functions.

Test and maintenance provisions will be specifically included in the control station designs. This is a good design practice and is, therefore, acceptable.

EPRI specifies that the control stations will have operator-adjustable lighting (between 10 and 50 foot-candles). EPRI indicates that this is consistent with the results documented in EPRI NP-5989, "Effects of Control-Room Lighting on Operator Performance, A Pilot Empirical Study."

The control station design will also include a list of material, spare parts, and references that are to be stored at or near the control station.

EPRI states that the areas that are continuously staffed will have an ambient noise level less than 60 dB(A), including the use of the emergency HVAC system. This is a positive goal and it should be obtainable. Additional discussion of noise is included in Appendix B to this chapter.

The passive Requirements Document specifies that continuously staffed areas will be provided with carpet and sound-absorbing walls and ceilings whose materials are selected for fire resistance and ease of upkeep, durability, freedom from generation of static electricity, and appearance. This is acceptable.

The staff's evaluation of the human factors aspects of this section is given in Appendix B to this chapter.

4.8 Control Panels

Section 4.8 of Chapter 10 of the passive Requirements Document states that the panels for the M-MIS will be designed in accordance with EPRI NP-3659. The location of equipment on a panel will be determined by the functions and tasks identified for the control station. Final arrangements will be determined after review in a full-scope simulator. The designer will provide a consistent approach to identifying the functional divisions of the panels and the components. The panels will be designed to facilitate future modification.

The design configuration drawings will include defined space for any required operator aids, which are also to be specified by the designer.

4.9 Requirements for Specific Control Stations

4.9.1 Main Control Room

Section 4.9.1 of Chapter 10 states that the design of the MCR will be consistent with EPRI NP-3659; EPRI specifies the basic requirement that the control stations be designed so that the tasks assigned to them can be performed. The designer will specifically identify paths to and from the MCR for specific tasks. Access by personnel to and from the MCR will not interfere with the activities of the control room operators.

EPRI states that the MCR will be designed to accommodate personnel other than direct operating staff. Access for NRC inspectors, utility management, personnel providing engineering assistance, personnel performing maintenance activities on MCR equipment, shift supervisor, and administrative support personnel will be included in the design of the MCR. Access will also be provided for fire brigades.

Two exits will be provided for MCR evacuation. Section 4.9.1.2.3 of Chapter 10 states that the remote shutdown stations will be accessible to operators evacuating the MCR from either of two independent exits without the use of security devices such as keys or key cards or electric power. The staff considers the remote shutdown panel to be vital equipment as defined by 10 CFR 73.2. Vital equipment is required by 10 CFR 73.55(c) to be located in a vital area, to which access from within the protected area requires passage through a physical barrier. All unoccupied vital areas are required by 10 CFR 73.55(d) to be locked and protected by an activated intrusion alarm system, and all points of personnel access to vital areas are required to be positively controlled. If the remote shutdown stations and the paths to it from the MCR will be located within the same or contiguous vital areas, the requirements of both Section 4.9.1.2.3 and 10 CFR 73.55 would not conflict. If not, some accommodation may have to be made with regard to Section 4.9.1.2.3.

The staff recognizes the importance of providing rapid access to remote shutdown panels. Provisions are included in 10 CFR Part 73 to accommodate the need for rapid ingress or egress of individuals during emergency conditions or situations that could lead to emergency conditions. Sections 4.9.3.6 and 4.9.3.8 of Chapter 10 contain requirements that should protect against radiological sabotage due to unauthorized actuation of the remote shutdown stations by limiting the functions that could be taken away from the MCR and by timely alert of the control room to enablement of a remote shutdown station.

However, the remote shutdown panel also needs to be protected from tampering that does not involve its actuation but could render it useless to operators if tampering is not discovered before the panel is needed in an emergency. In its letter of October 12, 1990, EPRI stated that it is unlikely that positive means such as physical locks would be compatible with the functions of a remote shutdown station. If safety considerations are found to outweigh positive means of controlling access to the remote shutdown panel, there

should be some mechanism to detect unauthorized access, such as tamper-safe cabinet door alarms to annunciate access to displays and controls at normally unmanned vital control stations. The staff will evaluate proposed resolutions of this concern during its review of an individual application for FDA/DC.

Section 4.9.1 of Chapter 10 specifies that a restroom will be provided for the operators and that kitchen facilities will be provided near the control area for the operators' comfort.

EPRI states that postaccident conditions in the MCR will be monitored using normal plant instrumentation. This capability will be consistent with RG 1.97. This is acceptable.

EPRI states that the operators' and supervisors' workstations in the MCR will be compact and computer driven.

EPRI states that where direct voice communication is not practical, a means to summon operators from other parts of the MCR will be provided.

EPRI states that space will be provided in the MCR for shift turnover operations and for emergency equipment needed for control room personnel. The designer will identify any such equipment as part of the design and the assumptions for the need.

In its RAI dated May 17, 1991, the staff requested a clarification of Figure 4.10-2 to include the computer room within the security boundary of the MCR. In its response dated July 22, 1991, EPRI stated that Section 4.9.1 requires that a computer room be included in the security boundary for the MCR only if the analysis of functions and tasks shows that the operators need ready access to the computer room. In the DSER for Chapter 10, the staff stated that it believed that there is sufficient operating experience that demonstrates that operators need ready access to the computer room during both normal and off-normal operations and recommended that the sample drawing be revised to include the computer room. The staff stated that it would require a vendor to include the computer room within the security boundary of the MCR and identified this as an vendor- or utility-specific item and stated that it would evaluate this requirement during its review of an individual application for FDA/DC. In a letter dated September 29, 1992, EPRI stated that the computer room should be included in the security boundary of the MCR only if the analysis of functions and tests required by Section 4.9.1 shows that the operators need ready access to it. The staff concludes that EPRI has provided adequate requirements to address the security concern regarding controlling access to the computer room.

4.9.2 Local Control Stations

Section 4.9.2 of Chapter 10 states that the local control stations will be considered part of the M-MIS and will be designed to the same criteria as the MCR.

EPRI states that normal operation will not allow the controls in the MCR to be overridden from a local station or actions to be taken that generate a false

display signal. The MCR operator may have a control that permits override or indication that control has been lost. This requirement allows designs that comply with current regulatory criteria and is, therefore, acceptable.

For local control stations not normally used, EPRI states that less stringent environmental conditions are acceptable. Minimum levels of communication capability and lighting will still be maintained. In the DSER for Chapter 10, the staff concluded that this was not a good design practice. Stations predicted to be used minimally can still be very important and must still have adequate environments so that local actions can be taken safely. Although this requirement does not violate NRC regulations, the staff stated that EPRI should reconsider its appropriateness and identified this as an open issue. In its letter dated September 29, 1992, EPRI explained that the intent of this requirement is to provide realistic limits and criteria for little-used local stations. Specifically, Section 4.9.2.2 requires that no condition be predicted to prevent the operators from carrying out the assigned tasks under accident as well as normal operating conditions. The staff concludes that this explanation is acceptable; therefore, this open issue is closed.

EPRI states that the stations will be designed for one-person staffing but will allow two.

EPRI specifies that inadvertent actuation of controls at the local control stations will be eliminated by design to the extent practical. In the DSER for Chapter 10, the staff concluded that this was a good design goal, but that EPRI had not provided detailed design guidance. It stated that EPRI should provide such guidance and identified this as an open issue. In its letter dated September 29, 1992, EPRI explained that Section 4.9.2.4 requires the designer to eliminate, to the extent practical, the inadvertent actuation of local control stations by design. In addition, Section 4.5 references guidance in EPRI NP-3659, which has a subsection devoted to "strategies for preventing accidental actuation control." The staff concludes that EPRI has adequately addressed guidance on the inadvertent actuation of controls at local control stations. Therefore, this open issue is closed.

Section 4.9.2.5 of Chapter 10 states that positive means will be provided to prevent unauthorized use of normally unmanned local control stations that could have serious consequences. EPRI states that this requirement would not apply to stations needed for emergency use or where it otherwise would not be practical. In the rationale portion of this section, EPRI states that the designer must balance the conflicting considerations of security and ease of operator actions in an emergency. Section 5.2.4.2 of Chapter 9 provides a mechanism for achieving this balance by requiring an evaluation to ensure that security restrictions for access to equipment are compatible with required operator actions during all normal and emergency modes of operation. The staff concludes that the provision to prevent unauthorized use is consistent with 10 CFR Part 73 and is, therefore, acceptable.

4.9.3 Remote Shutdown Control Stations

Section 4.9.3 of Chapter 10 states that remote (outside the MCR) shutdown capability will be provided to go to hot standby and remain there indefinitely and to reach cold shutdown in 72 hours if needed. As with the rest of the M-MIS, the designer will identify the functions and tasks to be performed at

the remote shutdown stations. Since conditions requiring evacuation are unpredictable, no particular evacuation scenario can be assumed for the remote shutdown stations.

The plant simulator will include the remote shutdown stations and permit simulated shutdown using these facilities.

EPRI indicates that the remote shutdown stations will be designed with the assumption that a reduced number of staff may be available for short-term operation, that is, to bring the reactor to hot standby, but that a normal crew will be available for subsequent operations. The arrangement of the remote shutdown station will be the same as that of the MCR. The remote shutdown stations may be used for normal operations to avoid disruptions in the MCR.

Section 4.9.3.6 of Chapter 10 requires the designer to evaluate the possible routine use of the remote shutdown station for some normal operations. It also states that for any controls added to a remote shutdown station to support normal operations, the designer will specifically consider the potential for increased vulnerability to insider sabotage based on the sabotage vulnerability analysis required by Section 5.2.2.1 of Chapter 9. The staff concludes that these requirements are consistent with those of 10 CFR Part 73 and are, therefore, acceptable.

Section 4.9.3.8 of Chapter 10 addresses the need to protect remote shutdown stations from unauthorized activations. The proposed method is to "annunciate" (in the MCR) any unauthorized actions that occur in the remote shutdown station. This section also requires designers to assess the annunciation (in the MCR) to ensure "adequate time for the operators and plant security to take action to prevent a serious accident." The staff concludes that this is acceptable.

EPRI states that the designer will identify reference material, and space will be provided in the design for this material at the remote shutdown station. The remote shutdown station will be functional regardless of a failure in the MCR and will be physically separated from the MCR. This is the intent of the NRC requirement in Section III.G of Appendix R to 10 CFR Part 50 for a remote shutdown station and is, therefore, acceptable.

4.9.4 Emergency Response Support Facilities

Section 4.9.4 of Chapter 10 states that the design of the emergency response support facilities will meet the requirements of 10 CFR 50.47(b) and Appendix E (IV) of 10 CFR Part 50, the criteria in NUREG-0696, "Functional Criteria for Emergency Response Facilities," and Section 8, "Emergency Response Facilities," of Supplement 1 to NUREG-0737.

EPRI specifies that the TSC will be within a 2-minute walk of the MCR. Personnel in the TSC will be able to view the MCR activities through windows or CCTV. The TSC may be used for normal operations as long as it remains fully available for emergency support.

EPRI considers the EOF to be outside the scope of the passive Requirements Document. Therefore, the staff will evaluate the acceptability of the EOF design during its review of an individual application for FDA/DC.

Section 4.9.4.1 requires that, in an emergency, it be possible to modify normal security boundaries (using defined procedures) so that personnel can move between the MCR and the TSC without crossing a security boundary. In its July 13, 1990, RAI, the staff commented that procedures for modifying security boundaries in an emergency must be consistent with 10 CFR 50.54(x) and (y). By letter dated October 12, 1990, EPRI stated that if the plant owner intends to modify the normal MCR security boundary in an emergency, EPRI expects that this would be covered by the technical specifications and a specific plant procedure provided, so that applying 10 CFR 50.54(x) and (y) would not be necessary. This response is acceptable, and the staff will evaluate the acceptability of the provisions taken to address modification to security boundaries during its review of an individual application for FDA/DC.

Section 4.9.4.3 of Chapter 10 states that the data available to the TSC and EOF will be the same as the data available in the MCR and will be displayed in the same format, insofar as practicable. The transmission of data to the TSC and EOF will be single failure proof.

EPRI specifies that processing capabilities will be provided for the TSC and EOF that do not place a burden on the plant process computer. In its RAI dated April 10, 1990, the staff asked EPRI to describe the data storage methods that are allowable. By letter dated July 23, 1990, EPRI responded that the designer will specify the detailed requirements and the passive Requirements Document will not specify or restrict any particular method. Therefore, the staff will evaluate this issue during its review of an individual application for FDA/DC.

4.10 Conclusions

The staff concludes that the requirements in Section 4 of Chapter 10 do not conflict with current regulatory requirements and are, therefore, acceptable. However, the requirements by themselves do not provide sufficient information to make a determination that a specific design application will be acceptable. Therefore, applicants referencing the passive Requirements Document will be required to demonstrate compliance with the guidance in the SRP or provide justification for alternative means of implementing the associated regulatory requirements.

5 DATA GATHERING, TRANSMISSION, AND PROCESSING REQUIREMENTS

5.1 Definition

Section 5.1 of Chapter 10 of the passive Requirement Document states that Section 5 provides the requirements defined by EPRI that are needed to ensure proper acquisition, processing, and distribution of the data for control, monitoring, and protection.

The functions addressed in this section are those required to

- provide adequate data quality
- provide sufficient data handling capacity
- meet the requirements without being restricted to particular equipment
- provide guidelines for appropriate data volume
- provide data requirements for the TSC, EOF, and operator aids

5.2 General Requirements

5.2.1 Architecture

Section 5.2.1 of Chapter 10 of the passive Requirements Document states that the designer will establish the data system structure. Items for the designer to consider include minimization of the plant wiring, support of the maintenance and test requirements, and single component failure. Hardwired or serial transmission will be allowed if needed for speed or diversity. The designer will establish the number of "layers" and the number of data paths to be used.

EPRI specifies that the plant data system will be designed to accommodate the use of standard devices using standard protocols for the serial and parallel connections. The data system will be redundant and separated to the same extent as the systems to which it is connected. Because of the potential magnitude of disruption to the M-MIS should data transmission cables be severed or otherwise rendered inoperable, special measures are necessary to guard against accidents and tampering, for example, use of conduit for data transmission cables. In the DSER for Chapter 10, the staff concluded that such requirements should be added to the passive Requirements Document or clearly identified if they already exist. This was identified as an open issue. In its letter dated September 29, 1992, EPRI stated that the existing requirements provide for segmentation of major functions and require redundancy and separation of safety and non-safety cables. EPRI has added Section 6.2.11.1.7, which requires that data transmission cables for the M-MIS meet the separation criteria in IEEE 384 (1981), Section 6. However, the staff concludes that this issue is dependent on specific passive plant design and will be resolved by the individual passive plant vendors during the staff's review of an application for FDA/DC. Therefore, this open issue is closed.

It is customary for licensed power reactor security computer systems to be independent of plant process computers. The protected area perimeter intrusion detection alarms and associated CCTV signals may require a dedicated data transmission system to provide the data transmission speed necessary to meet

the alarm assessment requirements of 10 CFR 73.55(h)(4)(i) and (ii). Section 5.2.1.1 of Chapter 10 contains nothing that would preclude such an independent data transmission system for perimeter intrusion alarms. However, the staff will evaluate the capability of the alarm assessment system to meet the requirements of 10 CFR 73.55(h)(4) during its review of an individual application for FDA/DC.

5.2.2 Design Process Requirements

Section 5.2.2 of Chapter 10 specifies that the data system will be consistent with the same design guidance as that for the rest of the M-MIS. The signals provided will be designed to support their use with a minimal use of special high-performance devices. The design process will include guidelines for determining data system characteristics. In the DSER for Chapter 10, the staff concluded that this section was acceptable, but that EPRI had not provided guidance for implementing the requirements and identified this as an open issue. In its letter dated September 29, 1992, EPRI stated that Section 5.2.2 provides sufficient guidance on the data system and any additional guidance would be an encroachment on the system design. The staff concludes that EPRI has provided adequate guidance on the data system characteristics. Therefore, this open issue is closed.

In the DSER for Chapter 10, the staff concluded that although the passive Requirements Document addressed the requirements of redundancy and diversity in the data system through reference to Sections 3 and 6 of Chapter 10, these sections did not adequately cover the need for requirements that specifically guard against the propagation of common-mode failures through the data system. This was identified as an open issue. In its letter dated September 29, 1992, EPRI stated that the requirements in Section 3.5.3.2 address the issue of common-mode failures and their effects in determining the architecture and choosing hardware and software designs for the complete M-MIS. These requirements are fully applicable to all software and hardware features in the M-MIS including the data systems. The staff concludes that Section 3.5.3.2 provides sufficient requirements to guard against the propagation of common-mode failures through the data system. Therefore, this open issue is closed.

5.2.3 Performance Requirements

Section 5.2.3 of Chapter 10 specifies that the multiplexers, data communication links, and network links will have sufficient (approximately 40 percent) performance margin under conditions of maximum stress and that the designer will evaluate failures and operator actions. The multiplexers will have reasonable expansion capability that would permit the utility to add functions in the future. In the DSER for Chapter 10, the staff noted that Section 5.2.3.2 of Chapter 10 stated that the expansion capability should be a minimum of 25 percent. However, Section 5.2.3.3 indicated that 30-percent expansion capability should be provided. These numbers were inconsistent and should be clarified. This was identified as an open issue. In its letter dated September 29, 1992, EPRI explained that the intent of this requirement is to specify the hardware spare capacity requirements for future use. This clarifies the inconsistency identified in the DSER. Therefore, this open issue is closed.

5.2.4 Reliability and Availability

Section 5.2.4 of Chapter 10 states that the data system will meet the reliability and availability of the system that uses the information. When redundant data paths and signal selection are used, the reliability model of the data path will include consideration of the failure rate and coverage provided by the selection device or algorithm. The reliability analysis must include consideration of cases where the selection algorithm or device will not detect all failures or data anomalies. The measurement of reliability and availability has been addressed in previous sections.

5.2.5 Signal Transport Delay

Section 5.2.5 of Chapter 10 states that the propagation time for multiplexed data will be analyzed to ensure stability in closed-loop control. The time delays will be considered to ensure that the end actions occur within the allocated time. Response degradation due to filters, sampling rate, analog-to-digital (A/D) conversions, and so forth, will be included in the propagation time.

As discussed in Section 5.2.1 of this chapter, some security functions require rapid information system response. The staff expects that the provision in Section 5.2.5.1 of Chapter 10 for an analysis to "demonstrate the prevention of significant degradation in performance of plant control and monitoring systems" will allow the designer the flexibility to provide a dedicated security data transmission and processing system if needed to meet NRC alarm assessment requirements.

EPRI states that operator control feedback will be determined by human factors analysis. The design will provide acknowledgement of a requested action to the operator within 0.25 second. The system will not introduce more than 1.5 seconds of delay to a display than is currently used in hardwired systems. This is a reduction in system capabilities from the hardwired systems currently used. In its RAI dated April 10, 1990, the staff asked EPRI to provide additional justification for this reduction. By letter dated July 23, 1990, EPRI responded that the designer will consider the time delay and will justify that the response time of the system is adequate. The staff will evaluate this concern during its review of an individual application for FDA/DC. The staff's evaluation of the human factors aspects of this section is given in Appendix B to this chapter.

5.2.6 Standardization

Section 5.2.6 of Chapter 10 indicates that the system will be standardized to the extent practicable.

5.2.7 Communication Protocols

Section 5.2.7 of Chapter 10 states that standard communication protocols will primarily be used in the design, as previously mentioned.

5.3 Data Gathering Requirements

Section 5.3.1 of Chapter 10 of the passive Requirements Document states that the designer should consider the following data signal characteristics for analog signals:

- accuracy
- resolution
- sample rate
- repeatability
- response rate
- safety classification
- range

The list is not complete. However, these characteristics do not conflict with regulatory requirements and are, therefore, acceptable.

EPRI states that the data-gathering process for discrete and pulse input signals will provide:

- noise immunity
- sufficient response
- voltage supply for dry contact input
- switch debounce where needed
- identification of failure modes

The signal provided will meet the most restrictive value of each requirement from all of the systems that use the signal.

Section 5.3.2 of Chapter 10 specifies that each data acquisition channel will contain noise filters to eliminate spurious trips. A/D conversion will be provided with aliasing filtering as needed. This is a good engineering design and is, therefore, acceptable.

5.4 Data Transmission

Section 5.4.1 of Chapter 10 of the passive Requirements Document states that all data on the plantwide data buses will have signal identification information associated with them. Where precise timing information is important, time information will be attached to the signal. Data will have a signal quality tag associated with them so that data errors can be tracked to the source of the problem.

Section 5.4.2 of Chapter 10 specifies that the multiplexing system will provide sufficient fidelity design so that random bit errors will not degrade the reliability of the systems. The goal provided in this section is that less than 1 of 1.0E+10 signal transmissions cause significant errors in the operator's displays. In the DSER for Chapter 10, the staff noted that using the EPRI estimate of 500 million signals per day, there would be a significant error every 20 days. It concluded that EPRI should clarify the definition of "significant" for this item and identified this as an open issue. In Revision 3, EPRI revised Section 5.4.2 to require the data transmission process to have sufficient inherent integrity and error checking to ensure that random errors in the process do not degrade the availability and reliability of the

systems and functions that utilize the data. This revision clarifies the definition of "significant" for reliability of the multiplexing system. Therefore, this open issue is closed.

5.5 Signal Processing

Section 5.5 of Chapter 10 states that the signal processing will ensure that the accuracy, precision, and rate of response are consistent with the applications. The data system will provide signal validity checks. The primary emphasis of this section is to ensure that the signal processing does not corrupt the signal. This is good design practice and is, therefore, acceptable.

5.6 Operator Aids

Section 5.6 of Chapter 10 states that any operator aids provided will include self-diagnostics to warn the operator whenever the system is malfunctioning or unavailable.

Section 5.6.1 of Chapter 10 states that the designers will provide a technical specification monitoring function that will have the capability to warn the operator when a limiting condition for operation (LCO) is being approached or violated. The system will automatically acquire the results of automatic testing that could affect an LCO and will automatically log all LCO violations. Because EPRI's general requirement is that all systems will receive the same level of V&V as the safety systems, this requirement is acceptable.

Section 5.6.2 of Chapter 10 states that the designer will provide the capability to monitor the status of the long-term and short-term availability of the reactor protection system (RPS) and safety systems.

Section 5.6.3 of Chapter 10 provides for features to aid in periodic surveillance testing, special testing, maintenance and inspection planning, and identifying potential equipment degradation. Monitoring and trending that identify the degradation of equipment before total failure will be implemented where possible.

Section 5.6.6 of Chapter 10 requires that the post-trip review data and information function provide data collection capabilities to perform post-trip reviews in accordance with GL 83-28.

5.7 Hardware

Section 5.7 of Chapter 10 describes the general hardware requirements specified by EPRI.

The passive Requirements Document specifies that module configuration by back plane wiring is preferable to on-card configurable switching. EPRI states that the accuracy, resolution, and speed of analog-to-digital (A/D) and digital-to-analog (D/A) convertors must be suitable for the application. The requirements in this section are acceptable. However, all A/D and D/A convertors are not acceptable for all applications and the staff will evaluate specific details during its review of an individual application for FDA/DC.

5.8 Conclusions

The staff concludes that the requirements in Section 5 of Chapter 10 do not conflict with current regulatory requirements and are, therefore, acceptable. However, the requirements by themselves do not provide sufficient information to make a determination that a specific design application will be acceptable. Therefore, applicants referencing the passive Requirements Document will be required to demonstrate compliance with the guidance in the SRP, or provide justification for alternative means of implementing the associated regulatory requirements.

6 COMMON SOFTWARE, HARDWARE, AND CONTROL REQUIREMENTS

Section 6 of Chapter 10 of the passive Requirements Document addresses the common software, hardware, and control requirements for the design, implementation, and installation of the M-MIS. All systems listed in Chapter 10 will meet these requirements. The staff's evaluation of the human factors aspects of this section is given in Appendix B to this chapter.

6.1 Common Software Requirements

6.1.1 Definition

Section 6.1.1 of Chapter 10 of the passive Requirements Document defines the M-MIS software to include all software and firmware required for the operation and maintenance of the plant. EPRI originally stated that the requirements did not necessarily apply to software used for design aids. In the DSER for Chapter 10, the staff concluded that it might be appropriate for some design aids to incorporate these criteria and stated that EPRI should include a requirement that the designer identify the design aids used and justify any that do not meet these requirements. This was identified as an open issue. In Revision 3, EPRI revised Section 6.1.1.2 to require that the M-MIS software include all software, firmware, and associated design aids required for operating and maintaining the plant. The staff concludes that this revision provides adequate requirements for software design aids and tools. Therefore, this open issue is closed.

6.1.2 Design Process

Section 6.1.2 of Chapter 10 states that safety-related software will meet the QA requirements of Appendix B to 10 CFR Part 50 as described in NUREG/CR-4640, "Handbook of Software Quality Assurance Techniques." In the rationale portion of this section, EPRI describes four basic categories of software.

The first category is application software, which includes the plant design calculations and plant operating software. The second category is the support software, which includes operating systems, compilers, assemblers, development stations, debuggers, editors, databases, mathematical subroutines, system libraries, and utilities. The third category is the test and maintenance software used to carry out testing, operation, and maintenance functions. The last category is the training software used for instruction and in simulators.

These descriptions are acceptable for use in the following discussions. The staff notes that the operation function listed in the third category applies only to the test and maintenance function and not to plant operation (which is included in the first category). The requirement for safety-related software to meet Appendix B to 10 CFR Part 50 is acceptable. However, the passive Requirements Document should be more specific in the listing of applicable criteria. In its RAI dated May 17, 1991, the staff asked EPRI to reference IEEE 828, "Standard for Software Configuration Management Plans," and NQA 2.7, "Quality Assurance Requirements of Computer Software for Nuclear Facility Applications," in Section 6.1.2 of Chapter 10. In its response dated July 22, 1991, EPRI stated that the M-MIS designer is expected to determine the applicable standards in effect and ensure that the design will comply with

them in meeting these requirements. However, in the DSER for Chapter 10, the staff concluded that more specific requirements (such as NQA 2.7) should be referenced and would provide additional guidance for the type of equipment described in the passive Requirements Document. This was identified as an open issue. In Revision 3, EPRI revised Section 6.1.2.1 to require that design of software comply with the following documents: (1) 10 CFR Part 50, Appendix B, (2) ASME NQA-2A, Part 2.7, (3) RG 1.152, and (4) ANSI/IEEE-ANS-7-4.3.2. This addition provides specific QA requirements for safety-related software. Therefore, this open issue is closed.

EPRI requires that a software life cycle be established to provide a systematic approach to the development, use, and operation of a software system. In its RAI dated April 10, 1990, the staff asked EPRI to add requirements for support software so that any errors and changes identified in the future will be reviewed for possible impact on installed software. By letter dated July 23, 1990, EPRI responded that it would revise this section. In the DSER for Chapter 10, the staff concluded that the proposed revision was acceptable and stated that it would confirm that it was acceptably incorporated into the passive Requirements Document. In its letter dated September 29, 1992, EPRI revised Sections 6.1.2.15 and 6.1.2.16 to add requirements for support software so that any errors and changes identified in the future will be reviewed for possible impact on installed software. The staff has verified the revisions have been made; therefore, this confirmatory issue is closed.

Section 6.1.2.3 of Chapter 10 describes the software life-cycle phases. Minimum requirements are listed. In the DSER for Chapter 10, the staff concluded that a configuration-management requirement similar to American National Standards Institute/Institute of Electrical and Electronics Engineers (ANSI/IEEE) 828-1983 should also be included and identified this as an open issue. In its letter dated September 29, 1992, EPRI explained that ANSI/IEEE 1042 is referenced in Section 6.1.2.18, and in a letter dated February 24, 1993, EPRI deleted the second sentence from the requirements of Section 6.1.2.1 to eliminate the part that is redundant to the first part of the third sentence. The staff agrees that requiring that software configuration meet the requirements of ANSI/IEEE 1042 provides adequate configuration-management requirements for software. In Revision 5, EPRI submitted a further modification to Section 6.1.2.1 that clarified that the SQAP for all but safety-related software will meet all requirements in this section, except for those in Appendix B of 10 CFR Part 50. The safety-related software will meet all the requirements listed in this section, including those in Appendix B of 10 CFR Part 50. Therefore, this open issue is closed.

EPRI requires the M-MIS designer to prepare a software requirement specification. The staff notes that this applies only to safety-grade and non-safety-grade software developed by the M-MIS designer. The staff will evaluate this specification during its review of an individual application for FDA/DC.

In Part I of the DSER (Section 2.2, "Software Integrity") for Chapter 10, the staff concluded that EPRI should consider the following changes in Section 6 of Chapter 10 of the passive Requirements Document:

- addition of requirements for the use of modern software development processes to reduce the chance of software errors, to find the errors that occurred, and to prevent unfound errors from causing unacceptable consequences

- the grading of the requirements in Section 6.1 with respect to different levels of importance to safety in a manner consistent with that for the sections identified for system integrity

- addition of requirements for building confidence in the software after the completion of the software development process for the functions ranked in the higher levels of importance to safety

The staff identified this as an open issue. In response to the first recommended change, EPRI added Section 6.1.2.22 in Revision 4 to require the M-MIS designer to use modern software development techniques and practices and to present examples of modern software development methods. In response to the second recommended change, EPRI added Section 6.1.8 in Revision 4 to provide software design process requirements for safety- and non-safety-related software. Section 6.1.2.1 was also changed in Revision 3 to clarify the SQAP for non-safety software. In a letter dated September 29, 1992, responding to the third recommended change, EPRI stated that requirements governing activities performed as part of the development process and not activities performed after the completion of the software development process must be adequate. The staff concludes that EPRI's revisions to address the first and second parts of this open issue are acceptable. For the third part, the staff will require that an applicant for an individual application for FDA/DC provide additional evidence that the final software product performs as intended. Therefore, this open issue is closed.

The passive Requirements Document specifies that the M-MIS designer will define the algorithms, equations, logic, and data operations.

EPRI requires that a software V&V plan be developed in accordance with ANSI/IEEE 730, "Software Quality Assurance Plans;" ANSI/IEEE 829, "Software Test Documentation;" and ANSI/IEEE 1012-1986, "IEEE for Software V&V Plan." The staff concludes that these standards are suitable for establishing QA plans, documenting testing, and establishing a V&V plan. The requirement in Chapter 1 of the passive Requirements Document to meet RG 1.152, "Criteria for Programmable Digital Computer System Software in Safety-Related Systems of Nuclear Power Plants," is an acceptable method for establishing a V&V plan and that guide, in addition to the standards listed above, is acceptable.

EPRI states that a software V&V report will be developed to provide the results of the V&V. The staff concludes that this is a standard requirement for good software design procedures and is, therefore, acceptable.

Section 6.1.2.8 of Chapter 10 states that the M-MIS designer will establish a coding standard so that each software module is formatted like every other module. The software will be designed with descriptive statements and comments incorporated. The staff concludes that this is good software design practice and is, therefore, acceptable.

EPRI states that the M-MIS designer will use code analysis to verify that the computer program correctly implements the design. The staff concludes that code analysis is a useful verification technique that does not conflict with current regulatory requirements and is, therefore, acceptable. However, it is not adequate by itself, and other methods of verification will be required to support a specific design application. The staff will evaluate this issue during its review of an individual application for FDA/DC.

EPRI states that the designer will provide user documentation. In the DSER for Chapter 10, the staff concluded that EPRI should reference sources of additional guidance, such as ANSI/IEEE 1063. This was identified as an open issue. In its letter dated September 29, 1992, EPRI explained that Section 6.1.7.2 references guidance in ANSI/IEEE 1063, "Standard for Software Users Documentation." The staff agrees that ANS'/IEEE 1063 provides adequate guidance on software user documentation. Therefore, this open issue is closed.

EPRI considers commercially available software acceptable if the purchasing organization performs acceptance tests or V&V on it. In the DSER for Chapter 10, the staff concluded that the general concept of using commercially proven software was acceptable. However, it stated that the requirement for acceptance testing was vague and that EPRI should clarify it. This was identified as an open issue. In Revision 3, EPRI revised the rationale of Section 6.1.5.8 to list the general guidance for developing and testing safety-related software. The revised section references RG 1.152, which endorses ANSI/IEEE ANS-7.4.3.2, and IEEE 1008, which provides guidance for acceptance testing. This revision clarifies the requirements for acceptance testing of commercially available software. Therefore, this open issue is closed.

The testing performed as required by Section 6.1.2.12 of Chapter 10 will be documented for later review by the licensee. After testing and/or verification, the software will be placed under configuration control. The staff concludes that this requirement is acceptable and will evaluate this issue during its review of an individual application for FDA/DC.

The purchase of software from clearinghouses that permit read-only access is allowed in the passive Requirements Document. In those cases, the clearinghouse will maintain the configuration control, but the purchasing group will be responsible for the accuracy of calculations, identification of software errors, and assessment of the impact of errors identified by other users. EPRI gives no information as to how this is to be accomplished. The staff is concerned that this may not be viable for safety system software and will evaluate this concern during its review of an individual application for FDA/DC.

Section 6.1.2.16 of Chapter 10 requires the software designer to maintain records of all commercially purchased software. It also requires the purchasing group to have a systematic method of informing users of changes required in the software. However, no information is provided in this section as to how a utility is to be informed of a subtier (e.g., an operating system or compiler) error by a software designer that is not required to comply with the requirements of 10 CFR Part 21, and may not be aware of all of the uses of the product. The staff concluded in the DSER for Chapter 10 that EPRI should

provide guidance on how this should be done and identified this as an open issue. In Revision 3, EPRI added "in accordance with IEEE 828 and 1042" to the rationale portion of Section 6.1.2.21. The staff concludes that this addition provides adequate guidance on the notification of software errors or modifications of commercially purchased software products. Therefore, this open issue is closed.

EPRI states that new software that needs to be developed will have to meet all of the requirements in Section 6.1.2 of Chapter 10. EPRI states that software configuration management will meet the guidelines specified in ANSI/IEEE 1042. In the DSER for Chapter 10, the staff concluded that EPRI should address how long-term configuration control will be maintained if several non-Class 1E vendors are involved and identified this as an open issue. In Revision 3, EPRI added Sections 6.1.2.19, 6.1.2.20, and 6.1.2.21 to require the plant designer to establish and maintain a software configuration management program for all software, commercially purchased or custom developed. The revised sections require that the plant designer establish a software configuration management plan in accordance with IEEE 828 and IEEE 1042. The staff concludes that this provides adequate guidance on long-term configuration control of software. Therefore, this open issue is closed.

In Revision 4, EPRI added Section 6.1.2.22, requiring the M-MIS designer to use modern software development techniques and practices and presenting examples of modern software development methods that should be considered by the M-MIS designer.

6.1.3 Software Design

Section 6.1.3 of Chapter 10 states that software will be developed using a top-down structured approach. The Yourdon methodology is the approach preferred by EPRI. Because this approach has been widely used and proven, the staff concludes that its use in the software design process is acceptable. However, it stated in the DSER for Chapter 10 that EPRI should clarify how or if this requirement will apply to all software or just to software developed by or specifically for the M-MIS designer. This was identified as an open issue. In Revision 4, EPRI revised the rationale of Section 6.1.3.1 to explain that the top-down structured design approach is applicable for all software developed by the plant designer or specified by the plant designer for the specific M-MIS design. The staff concludes that this clarifies EPRI's intent. Therefore, this open issue is closed.

EPRI states that software documentation will be developed along with the design. The software should be designed to reject out-of-bounds inputs. The staff concludes that this is a good engineering practice and is, therefore, acceptable.

The passive Requirements Document states that the software design will avoid convoluted software structure. In the DSER for Chapter 10, the staff stated that EPRI provided no guidance on how to determine if this requirement was satisfied and concluded that EPRI should provide such guidance. This was identified as an open issue. In Revision 3, EPRI revised Section 6.1.3.4 to require that the software development plan be in accordance with ANSI/IEEE ANS-7.4.3.2. The plan also must include the methodology for ensuring that the software is both auditable and testable during the design, implementation, and

integration phases. The staff concludes that these revisions provide adequate guidance on the convolution of software structure and the need for graded software. Therefore, this open issue is closed.

EPRI states that operating system software will not be modified by the downstream users. Section 6.1.3 of Chapter 10 discusses portability and upward compatibility when updated and new operating system software is released. In its RAI dated April 10, 1990, the staff noted that extreme care and control are needed in this area. There have been many examples of updated software that were not completely compatible. By letter dated July 23, 1990, EPRI agreed with the staff's comments, but did not propose any additional guidelines in this area.

Section 6.1.3.7 of Chapter 10 states that a hierarchical design structure and principles of modular design will be used in the software design.

In Section 6.1.3.8 of Chapter 10, EPRI states that the designer will identify assumptions and will specify how the program will behave if the assumptions are violated. In the rationale portion of this section, EPRI notes that fault-tolerant design allows the software to continue functioning in spite of failures. In the DSER for Chapter 10, the staff concluded that the statement about fault tolerance was acceptable, but the requirement did not specify a fault-tolerant system, only one in which the results of the faults are known. The staff concluded that EPRI should address how this information regarding fault-tolerance will be determined for commercial software and identified this as an open issue. In its letter dated September 29, 1992, EPRI explained that Section 6.1.2.12 of Chapter 10 requires the plant designer to perform extensive validation testing for commercial software. The staff agrees with this explanation. Therefore, this open issue is closed.

Section 6.1.3.9 of Chapter 10 states that a minimum of different types of support software will be used. In its RAI dated April 10, 1990, the staff asked EPRI to address the tradeoff between potential common-mode software errors and the problems associated with using diversity in the design. Section 4.5.6 of Chapter 10 requires the provision of diverse, manual backup controls for safety systems. Section 3.5.3 of Chapter 10 addresses common-mode failures in general. EPRI states that it believes it is prudent to invest the available resources in one set of software and require stringent design. The staff concludes that these requirements do not conflict with regulatory requirements and are, therefore, acceptable.

EPRI states that support software will be proven commercial software packages.

Programming will be done in a high-level language such as FORTRAN. Assembly languages will be used only for low-level routines. The software is to be designed to be as machine independent as possible. The staff notes that although these requirements will be difficult to enforce for the use of commercial packages and support software, they do not conflict with regulatory requirements and are, therefore, acceptable.

EPRI states that the M-MIS designer will provide a plan for giving software support to the utility, including operating systems and compilers. This is a necessary ingredient for software design and is, therefore, acceptable.

Section 6.1.3.14 of Chapter 10 states that the M-MIS functions of protection, control, alarm, and display will be based on digital technology. This technology will have the following characteristics:

- software that is capable of being verified and validated
- a final source program that will be readable from start to end
- self-supervision of control flow and data
- common software language
- no assembly languages for protection or control, except in those cases where they are required to meet certain constraints
- standard software structure in processors which provide reactor protection or safety system functions
- continuous-loop, noninterruptible design
- global variables that will be located in a common region

In its RAI dated April 10, 1990, the staff stated that the use of common software language may minimize the potential for errors if significant V&V efforts are focused on that program. Common-mode failures may be more likely with some of the possible microprocessor-based systems than with previous analog designs. Section 3.5.3 of Chapter 10 addresses common-mode failures in general. The staff's evaluation of EPRI's requirements addressing common-mode failures is provided in Section 3.5.3 of this chapter.

Section 6.1.3.15 of Chapter 10 states that comprehensive diagnostic programs will be performed at initialization. It requires that diagnostic routines be bypassed during maintenance; however, before the equipment is put back in service, diagnostics must be run. The staff concludes that this ensures that the diagnostic routines are back in operation as required.

Section 6.1.3.17 of Chapter 10 states that parameters that may change because of plant conditions will not be hard-coded so that the code would have to be recompiled to change a parameter. In the DSER for Chapter 10, the staff concluded that the passive Requirements Document should provide additional guidance for memory protection. For example, attempts at writing to protected memory should be specifically prevented. The passive Requirements Document should also specify restrictions on the acceptable range within which an operator may make a change. This was identified as an open issue. In Revision 3, EPRI revised Sections 6.1.3.17, 6.1.3.17.1, and 6.1.3.17.2 of Chapter 10, to require that the software performing protective or control functions for safety-related systems reside in protective memory. The revised sections also require the M-MIS designer to specify the restrictions on the range and to provide steps to protect against and/or alert the operator from making constant changes beyond the specified range. The staff concludes that these revisions provide adequate specific restriction guidance on memory protection. Therefore, this open issue is closed.

EPRI states that a function that is widely used will be installed as a utility module that can then be shared by other software modules.

Section 6.1.3.19 of Chapter 10 states that specific steps will be taken to limit the possibility of software viruses. In the rationale portion of this section, EPRI notes that this is a developing area and it is, therefore, premature to specify detailed requirements. Examples given by EPRI include physically limiting access to input devices, independent verification of the validity of the input, software checksum techniques, and bit-by-bit comparison with secure copies of the software. The staff notes that although this list of examples does not include the use of software programs contained in read-only memory as firmware, the passive Requirements Document does not exclude such designs.

Section 6.1.3.20 of Chapter 10 describes the database management requirements that the M-MIS designer will provide.

EPRI states that process control and monitoring data that are updated will be stored in random access memory. Data that are used to define the plant will be stored in nonvolatile memory with the capability for integrity checking. In the DSER for Chapter 10, the staff concluded that EPRI should clarify the requirement that redundant safety-related devices not use the same database. The staff interprets this requirement to mean that the redundant safety channels will use the same information but that the information will be located in separate physical modules. EPRI should clarify the requirement to ensure that no redundant safety channels share the same modules until the point of logic voting. This was identified as an open issue. In Revision 3, EPRI revised Section 6.3.2.6 of Chapter 10 to add requirements to ensure that no redundant safety channels share the same modules until the point of logic voting. The staff concludes that this revision clarifies the requirement for information use by redundant safety channels. Therefore, this open issue is closed.

EPRI states that the use of rotary buffers is to be avoided. The staff concludes that this is good engineering practice and is acceptable.

In its RAI dated April 10, 1990, the staff asked EPRI whether the use of expert systems was restricted or encouraged. By letter dated July 23, 1990, EPRI responded that the use of expert systems does not meet the proven-technology requirement but they may be used in the future. The staff concludes that the response is acceptable. However, there are no acceptance criteria for expert or artificial intelligence systems, and the staff considers such systems unacceptable for use in safety-grade systems until such criteria are established and approved. The staff does not believe that the expert systems have been demonstrated to be deterministic in nature and that they can be adequately verified and validated.

6.1.4 Performance Requirements

Section 6.1.4 of Chapter 10 states that the computer system will be designed with about 40-percent extra performance margin to account for future expansion and uncertainties. The performance of the system will be measured to verify the margin. The staff concludes that this is good engineering practice and is, therefore, acceptable.

The computer system will have on-line diagnostics and will have the capability to do both hardware and software checks.

6.1.5 Verification, Testing, and Qualification

Section 6.1.5 of Chapter 10 states that the designer will develop tools as needed to improve the quality and reliability of the software. In the DSER for Chapter 10, the staff stated that though the rationale portion of this section listed many tools, such as debuggers and test drivers, the requirement did not give any specific guidance or recommendations. The staff concluded that EPRI should consider providing such guidance and identified this as an open issue. In Revision 4, EPRI revised the rationale portion of Section 6.1.5.1 of Chapter 10 to group the tools into categories according to use. The staff concludes that this revision provides the necessary guidance. Therefore, this open issue is closed.

EPRI states that the tools will be developed to the same level as the delivered software. In the DSER for Chapter 10, the staff concluded that EPRI should clarify that this includes performing V&V on software tools used to develop safety software to the same level as the V&V performed on the safety software itself. EPRI requirements concerning software design aids and tools are also evaluated in Section 6.1.1 of this chapter.

The software tools will be maintained under configuration management control. EPRI recommends that the tools be coded in high-level languages if possible.

EPRI states that the testing will consist of both human and computer-based testing.

Section 6.1.5.6 of Chapter 10 states that the test program objectives will be to locate programming errors and validate that the software performs correctly. In the rationale portion of this section, EPRI states the need to perform a reasonable amount of testing to provide a sufficient degree of confidence that the software is correct. In the DSER for Chapter 10, the staff concluded that this requirement was acceptable. However, it stated that EPRI should define how much testing was "reasonable" and what was a sufficient degree of confidence. This was identified as an open issue. In Revision 3, EPRI revised Section 6.1.5.6 of Chapter 10 to add guidance for the plant designer regarding the amount of testing required and referenced IEEE 1012, 981.1, and 981.2. In its letter dated September 29, 1992, EPRI stated that it considers that requirements governing activities performed as part of the development process to be adequate, not activities performed after the completion of the software development process. The staff concludes that it will review an individual application for FDA/DC to ensure that this issue is adequately addressed by the vendor. Therefore this open issue is closed.

EPRI states that the testing personnel will not include the individual or group that developed the software. In the DSER for Chapter 10, the staff concluded that this requirement was acceptable. However, the level of independence should be at least through the first level of management. This is necessary to ensure that the testers and verifiers are removed from the time and cost constraints that are usually present in developing a product. EPRI's requirements concerning software verifiers are also evaluated in Section 3.1.2 of this chapter.

The tester will define the expected results before running the tests in order to avoid interpreting errors as correct results. The staff concludes that this is a good practice and is, therefore, acceptable.

EPRI states that the testability, calibration, and bypass requirements of IEEE 279, "Criteria for Protection Systems for Nuclear Power Generating Stations," will be supported in a software-based design.

EPRI states that all software and firmware related to complying with Section 3.6 testability requirements will be designed and qualified to the same level as the base program software. Multipurpose design models must be designed to provide an adequate representation of the plant for all intended purposes.

Section 6.1.5.11.2 of Chapter 10 states that dynamic testing of the plant controls will be performed using the actual design models and design parameters. The testing will ensure an adequate and stable response when the process parameters are varied. In the rationale portion of this section, EPRI states that varying the process design parameters 20 percent above and below nominal values will ensure adequate and stable response of the plant control design.

6.1.6 Availability and Reliability

Section 6.1.6 of Chapter 10 states that the planning of the V&V will include systematic QA activities.

EPRI requires the M-MIS designer to evaluate the diversity and redundancy needed for each system. The designer will consider the need for a hardwired backup to the computer-based system. In the DSER for Chapter 10, the staff concluded that diversity in the basic technology could probably eliminate the need for software diversity. In the rationale portion of this section, EPRI correctly expresses the NRC staff concerns that software-based safety systems may contain subtle failure modes that occur only under an obscure set of conditions that may involve environmental factors as well as internal hardware or software failures. If the software failures were random, a multidivision safety system might provide the redundancy to back up such failures. However, if standardized software were used throughout the system, common-mode failures could degrade the overall safety system operation. The staff concluded that the resolution of this concern might require that various forms of hardware and software diversity be incorporated into the system designs. The staff agreed with EPRI that there were many different methods for obtaining diversity. It concluded that the passive Requirements Document should require some specific level of diversity in safety systems and identified this as an open issue. This is part of the policy issue on defense against common-mode failures in digital I&C systems identified in SECY-93-087. The staff will review individual applications for FDA/DC to ensure that applicants of future plants will address this issue as it relates to I&C system design. Therefore, this open issue is closed.

Section 6.1.6.3 of Chapter 10 requires that a reliability evaluation be performed. However, there appears to be little agreement in the industry as to how best to accomplish the task. Although there are IEEE standards (such as ANSI/IEEE 982) that provide guidance in this area, the nuclear industry has

not widely accepted any one method. In the DSER for Chapter 10, the staff concluded that EPRI should specify guidance on acceptable methods to perform such a reliability evaluation and identified this as an open issue. In its letter dated September 29, 1992, EPRI stated that the requirements in Section 6.1.6.3 of Chapter 10 are intended only to ensure that the designer achieves high-quality software as part of the design effort. In addition, EPRI stated that software reliability will be provided by the quality of design and will be achieved by application of a rigorous design process. The staff acknowledges the difficulty in generating reliability and availability estimates for use in a probabilistic risk assessment for comparison with a safety goal. Therefore, the staff will review individual applications for FDA/DC to determine whether a reliability evaluation has been properly performed and this open issue is closed.

EPRI states that security protection will be provided to prevent unauthorized or inadvertent changes to software and data.

Section 6.1.6.5 of Chapter 10 requires that software-based safety systems satisfy the basic design principles of defense-in-depth, redundancy, separation, independence, and diversity. EPRI states that the specific details of how to accomplish this are to be examined during the design process. In the rationale portion of this section, EPRI notes that the staff has been concerned that the existing regulatory criteria do not address the special design requirements of computers.

NUREG-0493 is given as an example of guidance for performing a common-mode failures analysis. EPRI states that the test program objectives will be to locate programming errors and validate review in this area. In the rationale portion of this section, EPRI describes a method of carefully distributing diverse modules through the systems to avoid the need for extensive diversity. The staff concludes that this requirement is acceptable; it only requires that there be enough diversity in the safety systems to sufficiently minimize the potential for common-mode failures due to software problems. The staff will evaluate this issue when EPRI defines the level of diversity in safety systems, as discussed previously. In addition, the staff will evaluate the acceptability of a specific design during its review of an individual application for FDA/DC.

Although Section 6.1.6.5 of Chapter 10 refers to IEEE 603, "Trial-Use Standard Criteria for Safety Systems for Nuclear Power Generating Stations," there is no stated commitment to invoke it or its supplement, ANSI/IEEE 7-4.3.2, "Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations," (RG 1.152). The staff concludes that the commitment to RG 1.152 in Chapter 1 of the passive Requirements Document without comment is not acceptable because of the very different interpretations of this RG by the vendors. In its RAI dated April 10, 1990, the staff asked EPRI to discuss more fully the V&V commitments in this chapter. By letter dated July 23, 1990, EPRI stated that Section 6.1.2 contains the requirements for the design process, QA, and V&V that describe elements of a software design process that experience has shown will produce high-quality software. Section 6.1.2 recommends the use of such guidance documents as NUREG/CR-4640, ANSI/IEEE 730, and ANSI/IEEE 829. As discussed in Section 6.1.2 of this chapter, these documents provide guidance pertaining to QA and testing documentation that the staff agrees provide an acceptable approach

for QA. However, it does not consider these documents adequate to develop a V&V plan for safety software. EPRI concluded its response by stating that the standards and regulations regarding the design of computer-based controls are evolving and, therefore, it is inappropriate to include an exhaustive list of standards and regulations in the passive Requirements Document. As discussed in Section 6.1.2 of this chapter, the staff concludes that there is not enough information concerning software V&V in the passive Requirements Document to provide confidence that a vendor following this guidance would meet NRC requirements.

6.1.7 Maintainability and Serviceability

Section 6.1.7 of Chapter 10 states that the computer manuals provided by the vendors to the M-MIS designer should also be provided to the utility. The designer will provide the utility with all the documentation and tools required to maintain and modify the system.

In the DSER for Chapter 10, the staff identified software maintenance as an open issue. EPRI responded to the open issue by referencing IEEE 828 and 1042 in Section 6.1.7.3 in Revision 4. The staff believes that compliance with these two IEEE standards will ensure that the staff's software maintenance concerns are adequately addressed. Therefore, this open issue is closed.

6.1.8 Applicability of Software Requirements

In Revision 4, EPRI added Section 6.1.8 of Chapter 10 to specify its requirements for the software design process. EPRI requires that the M-MIS designer document the level assigned to each item of software. Requirements are specified for

- software used in safety-grade systems to accomplish safety functions
- software used in defense-in-depth systems to accomplish defense-in-depth functions
- software used to accomplish functions that are important to prevent personal injury, to meet plant performance objectives, or to avoid damage to major plant equipment
- software used to accomplish all functions other than those included above

6.2 Common Hardware Requirements

6.2.1 Definition

Section 6.2.1 of Chapter 10 of the passive Requirements Document defines the common requirements for the design, selection, and installation of the M-MIS hardware. These requirements apply to all M-MIS equipment.

6.2.2 General

Section 6.2.2 of Chapter 10 states that the control system will meet the common system and equipment requirements given in Section 7 of Chapter 10. The control system designs will be coordinated to provide smooth overall plant

control. EPRI states that strong interactions between reactor and power generation control require a coordinated design approach. The staff concludes that this is acceptable.

EPRI states that the control system monitoring will provide enough information to monitor performance trends.

Section 6.2.2.4 of Chapter 10 appears to repeat Section 6.2.2.2, but adds the requirement to maintain a 15-percent design margin.

EPRI states that upon reinstatement of power, the microprocessors and computers will "self-start," but the associated non-safety systems will not restart without operator action.

Instrument racks will be provided to facilitate maintenance and testing. Electrical wiring that needs external connections will be routed to a connector or terminal box to facilitate maintenance. Process sensing lines will not be run an excessive distance. The I&C equipment will be protected from potential hazards.

Instrument ranges will cover system operating ranges with margin. Normal operating conditions will usually be shown as mid-scale on the instrumentation. Analog signals will be either current inputs or differential voltage.

Section 6.2.2.13 of Chapter 10 states that instrument setpoint drift will not cause a violation of technical specification limits. The guidance that is provided includes RG 1.105, "Instrument Setpoints for Safety Related Systems:" Instrument Society of America (ISA) 67.15; and NUREG-0993.

EPRI states that all equipment will be inherently free from electromagnetic interference (EMI) and will not broadcast EMI, or shielding and isolation will be provided. This is acceptable. The staff has concerns in the EMI area because of the extensive use of electronics and computers. This is an area that requires emphasis during the design phase. In its RAI dated April 10, 1990, the staff indicated that EPRI should provide additional implementation guidance in Section 6.2.2.14 of Chapter 10. By letter dated July 23, 1990, EPRI stated that it is the designer's responsibility to demonstrate that this requirement is met. This may be done through shielding, restriction of communication devices, and testing of equipment. The staff concludes that this requirement does not violate NRC regulations and is, therefore, acceptable. However, restriction of communication devices should not prevent compliance with 10 CFR 73.55. The staff will evaluate issues associated with EMI during its review of an individual application for FDA/DC.

In the DSER for Chapter 10, the staff noted that an acceptable method of shielding and grounding is provided in IEEE 1050-1989, "IEEE Guide for Instrumentation and Control Equipment Grounding in Generating Stations," but this standard was not referenced in the passive Requirements Document. The staff concluded that this reference should be added and identified this as an open issue. In Revision 2, EPRI added IEEE 1050 to Section 6.2.9. The staff concludes that this addition provides adequate guidance on this issue; therefore, this open issue is closed.

EPRI states that the design will follow the guidance of DOD-HDBK-263, "Electrostatic Discharge Control Handbook for Protection of Electrical and Electronic Parts, Assemblies and Equipment." The staff agrees with the need to address electrostatic discharge in the design and concludes that this is an acceptable reference.

Section 6.2.2.16 of Chapter 10 states that M-MIS equipment located outside the containment will be capable of operating at temperatures between 40 °F and 120 °F. The I&C equipment will operate between 10 and 95 percent (noncondensing) humidity and at a maximum wet bulb temperature of 95 °F. The HVAC will be designed to provide the required environment.

Section 6.2.2.17 of Chapter 10 states that the equipment will be designed to operate properly when exposed to the expected variations of voltage and frequency. The margins specified by IEEE 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," ($\pm$ 10-percent voltage and $\pm$ 5-percent frequency) are included in this requirement. Power conditioners can be used where needed. Because the newer computer-based equipment is particularly sensitive, the staff considers surge withstand capability to be an area that requires increased emphasis during the design phase. EPRI states that the M-MIS will be designed to protect against failures resulting from transients, sags, surges, and noise inputs that may occur on the power input cabling from the external power system. The equipment will withstand electrical surges per IEEE C37.90.1-1989 and IEEE C62.41-1980. Sections 6.2.2.17, 6.2.2.18, 6.2.8.1, and 6.2.8.3 of Chapter 10 specify M-MIS equipment performance requirements when operating under a range of various power supply system variations and perturbations. In the DSER for Chapter 10, the staff concluded that EPRI should include a requirement that the designer develop a comprehensive interface specification between the M-MIS equipment and the external power supply systems that support it to ensure their compatibility and identified this as an open issue. In Revision 5, EPRI revised Section 6.2.2.17 to require that the plant designer develop a comprehensive interface specification between the M-MIS equipment and the external power supply systems to ensure their compatibility. The staff concludes that these revisions provide adequate requirements for the compatibility between M-MIS equipment and its external power supply systems. Therefore, this open issue is closed.

EPRI states that the designer will select equipment to minimize the need for power and HVAC and will provide for the replacement or repair of equipment that is not expected to last for 60 years as part of the design. Large I&C systems will be preassembled. The systems will be designed so that circuit cards and modules can be replaced during channel bypass without affecting system safety or reliability.

EPRI states that hazardous high-voltage areas will be labeled and shielded. This is acceptable. Specific conformance with the National Electrical Code or Occupational Safety and Health Administration regulations is outside the scope of this review.

Section 6.2.2.24 of Chapter 10 states that signal validation will be performed on all critical safety, control, and plant availability systems. In its RAI dated April 10, 1990, the staff asked EPRI to provide information concerning the methods intended by this requirement. By letter dated July 23, 1990, EPRI

responded that it is not the intent of the passive Requirements Document to provide such detailed design requirements. The staff will evaluate signal validation methodology during its review of an individual application for FDA/DC.

EPRI requires that the I&C equipment be designed to perform diagnostics and troubleshooting down to the lowest replaceable circuit board or module level. The diagnostics and testing will be designed so that the equipment can be tested in place. Power outlets will be provided at all local stations to facilitate maintenance with the appropriate precautions for electrical separation.

EPRI states that cabinets requiring maintenance inside an enclosure will have permanently installed lighting. This should be an aid to technicians performing diagnostics, repairs, and troubleshooting and is acceptable.

6.2.3 Computer Systems

Section 6.2.3 of Chapter 10 states that the hardware and software used throughout the plant will be standardized, as much as practicable. The designer will specify the design life of the equipment and provide for replacement as needed. All equipment will be maintained under configuration management control.

EPRI states that system testing and validation will be performed with the software and hardware totally integrated. Integrated testing and validation is required and will be documented. The staff concludes that this is a good design practice and is, therefore, acceptable.

Section 6.2.3.6 of Chapter 10 states that the M-MIS designer will evaluate and establish the need for redundancy and diversity of computer and peripheral equipment. In the rationale portion of this section, EPRI notes that the designer can perform a reliability analysis of the equipment and establish the need for redundancy and diversity. EPRI states that diversity may not be practical because diversity and standardization are sometimes mutually exclusive. A reliability analysis may not be an acceptable substitute for redundancy or diversity determinations in safety systems because of common-mode-failure concerns. The staff discusses further in Section 6.1.6 of this chapter.

EPRI states that the maintenance and development hardware will be supplied to the utility. The software and hardware licenses and warranties from the equipment vendors as well as documents and training credits obtained from vendors will be delivered to the utility.

Computer-based systems will be provided with a battery-backed clock. In the DSER for Chapter 10, the staff concluded that EPRI should consider including a requirement that the designer include an alarmed self-diagnostic feature on the clock update to verify battery conditions for systems supporting important functions. This was identified as an open issue. In Revision 3, EPRI revised Section 6.2.3.11 to require that the battery-backed calendar clock have a self-diagnostic feature to verify the battery condition for systems supporting

important functions. The staff concludes that this revision provides adequate requirements on alarmed, self-diagnostic features on the clock update. Therefore, this open issue is closed.

The auctioneered power supplies will be designed so that a power supply can be maintained and replaced without disrupting the system.

6.2.4 Switches

Section 6.2.4 of Chapter 10 applies to the various types of switches used through the plant, including proximity, contact, level, pressure, and position switches.

EPRI states that the designer will specify the accuracy and repeatability needs for each type of switch. Section 6.2.4.2 of Chapter 10 states that the designer should encourage the use of integrated logic for switches to simplify operator information.

The designer will select the use of wet or dry contacts. The designer will specify the design life for each switch, and the switch will be qualified for that lifespan.

The designer will consider intelligent logic design to detect switch failure. The designer must evaluate the tradeoffs of hardware/software complexity and switch failure modes versus the improved switch failure detection.

Switches will have provisions for testing, will be modular, and will be designed to permit position adjustment.

6.2.5 Sensors

Section 6.2.5 of Chapter 10 gives the requirements for various types of sensors including temperature, pressure, acoustic, optical, vibration, flow, neutron, radiation, level, current, and voltage sensors.

Section 6.2.5.1 of Chapter 10 states that the requirements on reactor coolant system (RCS) temperature detectors are given in Chapter 3, Section 2.2.11.

EPRI states that the designer will select the temperature-measuring devices. The designer will consider multiple sensors and specify their use in areas in which there may be thermal stratification. The staff concludes that this is a good design practice and is, therefore, acceptable.

The designer is responsible for specifying the accuracy and repeatability requirements and the calibration accuracy requirements.

The designer is allowed to use flow nozzles, averaging pitot tubes, and vortex shielding.

EPRI states that the designer will consider the use of temperature devices other than resistance temperature detectors and thermocouples, and mentions integrated circuit temperature devices specifically. EPRI only calls for

consideration of the devices and seems to imply that they are preferable but does not require them. The staff agrees that they should be considered and concludes that this is acceptable.

Section 6.2.5.8 of Chapter 10 states that sensors will be qualified for the full range of use.

EPRI states that sensors will have provisions for connection to microprocessor-based test equipment.

Section 6.2.5.10 of Chapter 10 states that design provisions will be included to minimize leaving sensor isolation valves aligned in the incorrect position after calibration. In the DSER for Chapter 10, the staff stated that EPRI did not provide specific guidance for this requirement and concluded that EPRI should provide such guidance. This was identified as an open issue. In Revision 3, EPRI revised Section 6.2.5.10 to provide additional guidance to minimize leaving sensor isolation values aligned in the incorrect position after calibration. The staff agrees with the methods now included in Section 6.2.5.10 to minimize leaving sensor isolation values aligned in the incorrect position after calibration. Therefore, this open issue is closed.

EPRI states that local logic modules will be replaceable and that sensors will have provisions for field and bench calibration.

Section 6.2.5.13 of Chapter 10 states that sensors will have no undetectable failure mode. In the DSER for Chapter 10, the staff concluded that EPRI should clarify this section to address the loss-of-oil concerns associated with capacitance-type pressure sensors. Excessive drift in a setpoint is a failure mode that may not be detected until calibration. The staff concluded that this requirement could be interpreted to require all sensors to have on-line diagnostics and calibration and identified this as an open issue. In Revision 3, EPRI revised Section 6.2.5.13 to require, to the extent practical, that the sensor have no undetectable failure mode. In addition, the revised section requires the plant designer to identify those sensors that cannot meet this requirement and justify why on-line diagnostics and calibration are not possible. The staff concludes that these revisions clarify the requirements for capacitance-type pressure sensors. Therefore, this open issue is closed.

EPRI states that diversity will include the principle of operation as well as function to minimize common-mode failures. The staff discusses diversity and common-mode failures in Section 6.1.6 of this chapter.

6.2.6 Isolation Devices

Section 6.2.6 of Chapter 10 requires that the digital and analog isolation devices meet RG 1.75, "Physical Independence of Electric Systems." Because it does not conflict with regulatory requirements, the staff concludes that this requirement is acceptable. However, additional requirements concerning the adequacy of isolation devices must be met before the staff can approve a specific design.

Section 6.2.6.2 of Chapter 10 states that fiber optic cable is an acceptable isolator. This section states that the designer must comply with RG 1.75 and General Design Criteria (GDC) 3, 21, 22, and 24 of Appendix to 10 CFR Part 50. The staff concludes that this is acceptable.

EPRI states that the designer is responsible for selecting the isolators. In the DSER for Chapter 10, the staff stated that the various types of isolators listed in Section 6.2.6.3 of Chapter 10 were generally acceptable but must be shown, by testing, to isolate under maximum credible fault conditions in accordance with IEEE 279. It stated that the NRC minimal acceptance review criteria for isolation devices were the following:

- For the type of device used to accomplish electrical isolation, the specific testing performed to demonstrate that the device is acceptable for its application will be described. This description should include elementary diagrams when necessary to indicate the test configuration and the application of the maximum credible faults to the devices.

- Data will be supplied to verify that the maximum credible faults applied during the testing were the maximum voltage and current to which the device could be exposed, and the method used to define the maximum voltage and current will be documented.

- Data will be supplied to verify that the maximum credible fault was applied to the output of the device in the transverse mode (between signal and return) and other faults, such as open and short circuits, were considered.

- The pass/fail acceptance criteria will be established before testing and will be documented.

- The seismic and environmental qualification for each device will be demonstrated.

- The measures taken to protect the safety systems from electrical interference (e.g., EMI, electrostatic coupling, and crosstalk) will be described.

- The isolators used to separate safety systems from non-safety-grade systems shall be powered by the Class 1E power supply.

The staff concluded that the passive Requirements Document should be revised to list these criteria and identified this as an open issue. In Revision 3, EPRI revised Section 6.2.6.1 as requested by the staff to add the minimal acceptance review criteria for isolation devices. The staff concludes that this revision provides adequate guidance on minimal acceptance review criteria for isolation devices. Therefore, this open issue is closed.

EPRI states that the analog isolators will not degrade the accuracy of the system below acceptable levels and will not degrade significantly over time. The devices will provide at least 80 dB common-mode rejection.

6.2.7 Valves (I&C Features)

Section 6.2.7 of Chapter 10 states that motor-operated valves will have the valve control and position indicators separate from the logic initiators. EPRI states that motor operators will be from standard design.

Section 6.2.7.3 of Chapter 10 specifies that the Class 1E motor-operated valves will have their thermal overloads set as stated in RG 1.106, "Thermal Overload Protection for Electric Motors on Motor-Operated Valves," and IEEE 741. This is consistent with the requirements in Section 6.5.2 of Chapter 11 of the passive Requirements Document.

Section 6.2.7.3 of Chapter 10 also specifies required characteristics for motor-operated valves. EPRI states that all motor-operated valves will meet the requirements of NRC Branch Technical Position (BTP) ICSB 18 (PSB), "Application of the Single Failure Criterion to Manually-Controlled Electrically-Operated Valves." EPRI states that the number of valves that require power removal in order to meet the single-failure criterion should be minimized. EPRI requires that this provision be limited to only those situations where design of the piping system to eliminate the need to remove power from the valve would result in a less safe design than the design that requires power removal.

Section 6.2.7.4 of Chapter 10 of the passive Requirements Document states that power-operated valves will have local position indication if stem position is hidden or does not indicate valve position, and manual valves will have security locking devices when needed. In Revision 3, EPRI added a sentence to Section 4.4.4 of Chapter 10, allowing manual valve remote position indication, if required.

EPRI states that the designer will determine the critical valves that require external and/or internal leakage monitoring. The staff concludes that this is acceptable.

EPRI states that diagnostic, calibration, and any special tools required for troubleshooting will be provided to the utility.

Failure modes will be selected to allow the plant to remain at power. The staff will evaluate the failure modes selected on a case-by-case basis during its review of an individual application for FDA/DC.

6.2.8 I&C Power Supplies

Section 6.2.8 of Chapter 10 states that the M-MIS equipment will function when exposed to the expected variations of input voltage and frequency. The staff discusses this topic in greater detail in Section 6.2.2 of this chapter.

Section 6.2.8.2 of Chapter 10 states that the M-MIS will operate over a dc input voltage range, taking into account battery voltage variations within design limits. In the DSER for Chapter 10, the staff stated that the M-MIS equipment should be designed to operate properly at a maximum voltage and at a minimum voltage equal to the battery final discharge voltage minus the voltage drop from the battery to the M-MIS equipment terminals during worst-case system loading. This was identified as an open issue. In its letter dated

September 29, 1992, EPRI stated that Section 6.2.8.2 adequately addressed this issue. After reevaluating EPRI's requirements the staff agrees that the requirements in Section 6.2.8.2 provide adequate guidance on voltage design of batteries and dc systems. Therefore, this open issue is closed.

Section 6.2.8.3 of Chapter 10 states that surge protection features on the power inputs will be designed and tested to withstand the worst-case surge limits of appropriate standards, such as IEEE C62.41-1980. In the DSER for Chapter 10, the staff concluded that EPRI should specify the limits that apply to this section and identified this as an open issue. In its letter dated September 29, 1992, EPRI explained that the limits of IEEE C62.41-1980 are specified in Section 6.2.8.3. After reevaluating EPRI's requirement, the staff agrees that Section 6.2.8.3 contains adequate guidance to ensure that surges will not cause damage to the power supply or equipment connected to the power supply. Therefore, this open issue is closed.

EPRI states that overcurrent protection will be provided and that subsystem dc supplies will be provided with overvoltage and overcurrent protection.

EPRI states that power source input connection points will be within an enclosure and readily accessible. Direct current (dc) power within an equipment bay will be supplied by dc power supplies within the same bay. Cables from the power supplies will be sized for the loads.

Section 6.2.8.9 of Chapter 10 states that non-Class 1E equipment will be connected to non-Class 1E power supplies and distribution systems and will not be connected to Class 1E power supplies and distribution systems. The staff concludes that this is consistent with the guidance of RG 1.75 and the isolation requirements of GDC 17 and is, therefore, acceptable.

EPRI states that if any part of a function is powered from an uninterruptible power supply bus, the entire function will be powered from the same bus.

Section 6.2.8.11 of Chapter 10 states that display, local indicators, and equipment in instrument loops requiring power will have an unambiguous means of indicating loss of power. The staff concludes that this is acceptable.

EPRI states that if equipment needs power other than that normally supplied, it will be supplied with its own power supply.

6.2.9 Grounding

Section 6.2.9 of Chapter 10 states that grounding for all M-MIS equipment will meet the requirements in IEEE 1050, "IEEE Guide for Instrumentation and Control Equipment in Generating Stations."

Section 6.2.9.1 of Chapter 10 states that protective power grounds will be routed separately from signal grounds. Means will be provided to ground all equipment. A protective ground will be provided for all cabinets where the operating voltage is more than 50 volts. The staff concludes that these requirements are consistent with grounding requirements pertaining to personnel hazard safety and reduction in electromagnetic interference and ground currents in signal ground wires.

EPRI states that the grounds will be connected so that one piece of equipment may be disconnected at a time without disconnecting other equipment. The staff concludes that this is a good design practice and is, therefore, acceptable.

EPRI states that a portable ground cable will be provided to ground out-of-service electrical equipment. Cabinet grounding will be provided.

The passive Requirements Document states that power supply common returns will be provided and grounded at one point to prevent ground loops. The staff concludes that this requirement to ground at one point is acceptable. However, in the staff's opinion, this is a method for reducing ground-loop problems and may not prevent ground loops, contrary to the rationale portion of this section.

Section 6.2.9.2 of Chapter 10 states that wiring shields will be grounded separately from the circuit modules so that the modules can be removed without ungrounding the shield connection. In the DSER for Chapter 10, the staff concluded that this requirement was acceptable if the designer could demonstrate that the configuration is qualified for its electromagnetic interference/radiofrequency interference (EMI/RFI) environment. The staff stated that EPRI should clarify this section to reflect this condition and identified this as an open issue. In Revision 2, EPRI revised Section 6.2.9 to specify that grounding for all M-MIS equipment meet the requirements in IEEE 1050. The staff concludes that this addition specifies adequate EMI/RFI considerations for wiring shields. Therefore, this open issue is closed.

EPRI states that analog signals will be grounded at a single point. Instrumentation cable shields will be grounded at only one location. Shields will be terminated on terminal blocks adjacent to the signal wires. In the DSER for Chapter 10, the staff stated that Section 6.2.9.2 of Chapter 10 provided guidance on shield-grounding methods to minimize leakage between signals and shields; however, it did not reference specific grounding standards. The staff concluded that the passive Requirements Document should reference these standards and identified this as an open issue. In Revision 2, EPRI revised Section 6.2.9 to specify that grounding for all M-MIS equipment meet the requirements in IEEE 1050. The staff agrees that IEEE 1050 provides adequate guidance on grounding. Therefore, this open issue is closed.

EPRI states that there will be sufficient isolation ground between system grounds to prevent ground loops. Instrument grounds will be designed to minimize the effects of common-mode voltage levels. The signal connections between systems will have isolators. In the DSER for Chapter 10, the staff concluded that EPRI should clarify if this meant qualified isolators between systems within a safety channel and identified this as an open issue. In its letter dated September 29, 1992, EPRI stated that qualified isolators between systems within safety channels are not a requirement. After reviewing EPRI's clarification, the staff agrees that EPRI's grounding requirements are adequate. Therefore, this open issue is closed.

EPRI states that the signal grounding scheme for assemblies, subassemblies, and subsystems will use a "branching" scheme to minimize ground loops. Instrument ground connections will be corrosion resistant. The communication systems will not use the instrument ground bus.

6.2.10 Electrical Penetrations and Seals

Section 6.2.10 of Chapter 10 states that the I&C signals will not be degraded if penetration connectors are used. Splices are permitted, but are to be avoided, especially for fiber optic cables. Penetrations will be provided with spare connectors. The effects of EMI/RFI from other electrical penetrations will be considered in the design of the I&C penetrations.

6.2.11 Cables, Fiber Optics, and Raceways

Section 6.2.11 of Chapter 10 states that the designer will determine the need for spare conductors. The cables will be color coded. Conductors will be identified in accordance with Insulated Power Cable Engineers Association (IPCEA) 5-61-402. Twisted shielded pairs will be color coded in accordance with ANSI C96.1.

EPRI states that the designer will establish internal panel wiring standards in accordance to the color and identification numbering standards established for the entire plant.

In Revision 4, EPRI added Section 6.2.11.1.7 to Chapter 10 to clarify the separation requirements for data transmission cables. As stated in Section 1.4.2 of this chapter, the staff will evaluate this matter during its review of an individual application for FDA/DC.

Section 6.2.11.2 of Chapter 10 states that low-level control and instrumentation cable tray systems will be separated from the power cable tray system to minimize any interference of signals. EPRI states that this can be accomplished by grounding metal trays and providing air space separation. Power cables will be shielded, and the shield will be grounded.

Section 6.2.11.2 of Chapter 10 states that internal panel wiring will follow the following separation criteria:

- Ac and dc circuits will be separated.
- Instrumentation wiring will be separate from control or power wiring.
- Separation between Class 1E circuits and non-Class 1E circuits and redundant Class 1E circuits will be in accordance with IEEE 384-1974, RG 1.75, and BTP CMEB 9.5-1.
- There will be 6-inch separation between redundant channels or that which can be established by analysis.
- Conduit openings will be sealed.

EPRI states that the minimum bend radius of the fiber optic cable will be considered in designing the cable raceway. EPRI requires the designer to install spare fiber optic cables during construction based on the cost, complexity, and effort required to replace damaged cables, and the potential that damaged cables may lead to equipment or plant downtime.

6.2.12 Field Termination and Splices

Section 6.2.12 of Chapter 10 states that field terminations and splices will be positively secured while being easy to disconnect. EPRI states that field terminations and splices will comply with RG 1.75. The staff concludes that this is acceptable. Although the staff agrees with the rationale portion of this section that the designer has the option to consider termination methods other than ring lugs, the disadvantage of time-consuming field wire terminations with ring lugs should not be considered an equal tradeoff with the security of termination that they provide. Terminals will be identified.

6.3 Common Control System Requirements

6.3.1 Definition

Section 6.3.1 of Chapter 10 of the passive Requirements Document defines the common requirements for the design, selection, and installation of specific M-MIS control systems. EPRI states that all control systems required for operation and maintenance of the plant are included in the scope of this section, which addresses the design, performance, availability/operability, testability and qualification, and maintainability and serviceability requirements for this equipment.

6.3.2 Design Requirements

Section 6.3.2 of Chapter 10 states that all safety-related control system software will be retained in programmable read-only memory or on battery-backed read-only memory chips. Constants, such as setpoints, will be maintained in nonvolatile memory.

EPRI states that software interface will be discouraged between safety and non-safety systems but will be allowed if the designer can demonstrate that isolation is provided.

When evaluating stability and response rates, EPRI specifies that the designers should consider the following:

- plant and equipment nonlinearities that change with the operating point (valves, plant processes, etc.)
- time delays
- the effect of sample rate and resolution
- the effect of sensing subsystem components
- the effect of environmental extremes and process conditions

The designer will define and design the systems that will remain in the known safe state following restoration of power after a loss of power. The staff concludes that although this requirement is acceptable because it should prevent changing the system from the known safety state, it will evaluate the safe state defined for specific systems during its evaluation of an individual application for FDA/DC.

6.3.3 Performance Requirements

Section 6.3.3 of Chapter 10 specifies that digital control algorithms will include an antiwindup feature when needed. Transfers between manual and automatic control will be bumpless.

The settings for all control parameters will provide a resolution and accuracy of the control system. EPRI states that experience has shown that 1 percent or better of the full-scale adjustment range is considered to be acceptable. The staff concludes that this is generally acceptable; however, the designer should verify that there are no conditions for which more stringent requirements are warranted. The staff will address this issue during its review of an individual application for FDA/DC.

Section 6.3.3.4 of Chapter 10 provides requirements to ensure that the designer will specifically address aliasing problems with sampled data. However, the reference to EPRI contractor work to establish the requirements in this section needs to be completed. The staff concluded, in the DSER for Chapter 10, that EPRI should update or clarify the requirements for this section and identified this as an open issue. In Revision 3, EPRI revised Section 6.3.3.4 to clarify the requirements that address the aliasing problems. This revision clarifies the requirements for signal reconstruction. Therefore, this open issue is closed.

Section 6.3.3.5 of Chapter 10 originally indicated that computers that have direct protection or control functions will not have interrupts. Section 6.3.3.6 states that functions are to be prioritized so that a needed protective function will interrupt a monitoring function if necessary. In the DSER for Chapter 10, the staff stated that these sections appeared to conflict and concluded that EPRI should revise the passive Requirements Document to more clearly state under what conditions interrupts are allowed. In general, the staff considers that the use of interrupts in safety systems should be avoided. This was identified as an open issue. In Revision 3, EPRI revised Sections 6.3.3.5, 6.3.3.6, and 6.3.3.7 to provide additional guidance on interrupts. The staff agrees with this revision, which allows interrupts if they improve system reliability or are required to perform a protective function or other functions promptly. Therefore, this open issue is closed.

EPRI states that the designer will consider the startup testing requirements and will design the I&C system to minimize special setups and equipment.

6.3.4 Availability/Operability

For non-safety systems, Section 6.3.4 of Chapter 10 stipulates the use of reliable components instead of redundant equipment to achieve reliability. In the DSER for Chapter 10, the staff concluded that although this requirement was not in violation of NRC requirements, a requirement encouraging redundancy to reduce challenges to safety systems in addition to the use of reliable components would be more appropriate. The staff concluded that EPRI should consider such a requirement and identified this as an open issue. In Revision 4, EPRI added Section 6.1.8 to Chapter 10 to require the applicability of

software requirements, and revised Section 6.1.2.1 to clarify the SQAP for non-safety software. The staff concludes that this addition and this revision provide adequate requirements for redundancy of safety systems. Therefore, this open issue is closed.

EPRI states that signals from other systems may be used in non-safety control systems to enhance reliability. In addition, the control function may be redistributed to another computer upon failure of one computer, thereby providing some redundancy. The staff concludes that this requirement is acceptable if it applies only to non-safety systems. Safety systems will have to be evaluated to ensure that implementation of this EPRI requirement does not violate NRC regulations.

The I&C system will be a distributed system with local control. Central computers will be used only for monitoring. The staff concludes that this configuration is preferred and that it is, therefore, acceptable.

6.3.5 Testability and Qualification

Section 6.3.5 of Chapter 10 states that the designer will establish the criteria for determining which of the redundant systems will be used.

6.3.6 Maintainability/Serviceability

Section 6.3.6 of Chapter 10 states that the maintenance interface will provide the capability to access the diagnostic, calibration, and other maintenance aids. All system inputs and outputs will be accessible. Maintenance activities will be indicated in the MCR.

6.4 Conclusions

The staff concludes that the requirements in Section 6 of Chapter 10 do not conflict with current regulatory requirements and are, therefore, acceptable. However, the requirements by themselves do not provide sufficient information to make a determination that a specific design application will be acceptable. Therefore, applicants referencing the passive Requirements Document will be required to demonstrate compliance with the guidance in the SRP or provide justification for alternative means of implementing the associated regulatory requirements.

7 OVERALL PLANT, REACTOR, AND REACTOR COOLANT SYSTEMS M-MIS REQUIREMENTS

7.1 Purpose and Scope

Section 7.1 of Chapter 10 of the passive Requirements Document states that Section 7 provides the requirements for the M-MIS that will monitor and control the overall power production of the plant as well as the reactor and RCS.

EPRI specifies that the M-MIS for the specific systems discussed in this section will be integrated and coordinated with the M-MIS for other plant systems.

7.2 General Requirements for Overall Plant, Reactor, and Reactor Systems Group M-MIS

Section 7.2.1 of Chapter 10 of the passive Requirements Document describes how the initial allocation of the functions of the M-MIS will be distributed. EPRI states that the M-MIS will provide for the monitoring and control of the overall process of producing energy in the reactor core to the delivery of electricity to the grid and removal of waste heat. The M-MIS will monitor neutron flux and adjust the reactivity. Core outlet temperatures (PWRs) will also be monitored. The M-MIS will monitor and control the RCS pressure. The M-MIS will also monitor and control the chemical and volume control and reactor water cleanup systems. Chemistry will be monitored using the process sampling system, and reactor coolant leakage will be monitored using the reactor coolant leak detection system. The M-MIS will monitor and control removal of reactor core heat and monitor and control the steam production process.

Section 7.2.2 of Chapter 10 states that the physical boundaries of the system are defined in Chapters 3 and 4 of the passive Requirements Document. The boundaries for the M-MIS will be the same as those for the primary systems and will also include the M-MIS hardware and software.

Section 7.2.3 of Chapter 10 states that the designer will use consistent monitoring and control design strategies. The staff concludes that this is acceptable, but notes that this consistency should not result in the elimination of the consideration of diversity.

Monitoring and control of systems for startup and shutdown will be included in the M-MIS. Local operation is preferred over MCR operation when possible. EPRI states that automation of the functions to reduce operator burden is allowed, if justified, but is not encouraged because automation would increase the complexity of the plant.

During normal operations, the M-MIS that control this group of systems will be automated or will otherwise provide for continuous operation without repeated operator tasks. Occasional tasks will not require automation.

EPRI states that systems will be automatically reconfigured when needed for personnel or equipment protection or when required because of safety system actuation. In the DSER for Chapter 10, the staff stated that Section 7.2.3.3

did not address the selection process for determining when automation or manual control is to be used and concluded that EPRI should provide such guidance. This was identified as an open issue. In its letter dated September 29, 1992, EPRI stated that adequate guidance for selecting automatic or manual control is provided in Sections 3.1.3.3.3, 3.4.3, and 7.3.3 of Chapter 10. The staff concludes that these sections do not provide adequate guidance. However, in a letter dated December 31, 1992, the staff identified an additional issue concerning the lack of requirements for equipment that performs automatic reconfiguration. In response to this concern, EPRI added Section 6.1.8 in revision to provide software requirements for the equipment that performs the safety functions. The staff concludes that the added section provides adequate software requirements. In addition, the requirements issue for the system and equipment is part of the policy issue on regulatory treatment of non-safety systems for passive reactor designs. As described in SECY-94-084, this issue will be fully addressed by future passive plant applicants for FDA/DC. Therefore, the staff concludes that this open issue is closed.

EPRI states that return of a system to its initial configuration after a reconfiguration will normally require manual initiation but could be automated when justified. Any automatic return to an original configuration will require operator notification.

EPRI specifies that the M-MIS will normally be tested on line only at the direction of the operator. The testing should be automated after it is initiated by the operator. Other testing requirements are evaluated by the staff in Section 3.6 of this chapter.

Section 7.2.4 of Chapter 10 states that the M-MIS designer will integrate and coordinate the systems covered by this section. As noted in the rationale portion of this section, the passive Requirements Document specifically avoids providing any specific guidance as to how this is best accomplished. Thus, the M-MIS designer is not constrained by current practice. The staff concludes that this is acceptable. However, the staff notes that the passive Requirements Document does specifically state that EPRI prefers proven technology.

The staff's evaluation of the human factors aspects of this section is provided in Appendix B to this chapter.

7.3 Overall Plant M-MIS

Section 7.3.1 of Chapter 10 states that the M-MIS functional goal is to efficiently and effectively deliver electricity to the grid. Section 7.3.2 of Chapter 10 originally stated that the M-MIS would enable the utility load dispatcher to adjust reactor power "directly." In its response dated July 23, 1990, to a RAI, EPRI stated that the intent of this requirement is to allow the load dispatcher to change power level within a range selected by the operators. In the DSER for Chapter 10 the staff concluded that this section could be misinterpreted to allow the manipulation of reactor controls by personnel other than licensed unit operators. Controls that directly affect the power of the reactor can only be manipulated by licensed operators. Apparatus and mechanisms other than controls that directly affect reactor power can be manipulated only with the consent and knowledge of the licensed

operator at the controls. Economic power generation systems in use today allow the utility load dispatcher to initiate reactor power changes, but these changes are actually made by the digital data acquisition and control system (DDACS). The DDACS controls the power change rate by "manipulating" the reactor controls. This system will be under the cognizance of the licensed unit operator at all times, and power changes by the load dispatcher will first be approved by the licensed unit operator. The staff concluded that this section should be revised to reflect that the utility load dispatcher cannot "directly" adjust plant power and identified this as an open issue. In Revision 3, EPRI deleted the word "directly" from Section 7.3.2 for clarification. EPRI also revised this section to require that the M-MIS provide the capability for each power change to be approved by the licensed unit operator before the change being effected. The staff concludes that these revisions clarify the operational role of a load dispatcher. Therefore, this open issue is closed.

Section 7.3.3 of Chapter 10 states that the overall M-MIS will be coordinated with the M-MIS of other systems. Section 7.3.4 of Chapter 10 states that the M-MIS will provide the capability to monitor the functions of all major components and the plant heat balance and to determine if a change in plant operation setpoints would provide more efficient operation. The staff concludes that this is acceptable.

7.4 Neutron Monitoring System M-MIS

Section 7.4 of Chapter 10 states that the neutron monitoring M-MIS will provide the information needed to determine if the amount, rate of change, and distribution of fission energy are correct. The designer will determine the design of the monitoring strategy. The staff will evaluate this M-MIS during its review of an individual application for FDA/DC.

7.5 BWR Rod Control System M-MIS

Section 7.5 of Chapter 10 states that the M-MIS for the BWR rod control system will provide the monitoring and control necessary to position the control rods within acceptable limits. EPRI states that the designer will determine which functions of the normally manually controlled shutdown and startup control should be automated.

EPRI states that during the design process, the functions and tasks involved in scram time testing will be analyzed, and the need for automation to provide accurate and reliable results will be determined.

7.6 PWR Rod Control System M-MIS

Section 7.6 of Chapter 10 states that the PWR rod control system will provide the monitoring and control necessary to position the control rods within acceptable limits. EPRI requires that the designer evaluate any automated actions needed for the normal startup and shutdown functions.

7.7 BWR Reactor Coolant System M-MIS

Section 7.7 of Chapter 10 states that the M-MIS for the BWR RCS will provide the monitoring and control necessary to

- adjust the reactivity
- maintain the RCS pressure within limits
- maintain RCS inventory
- remove core energy during normal operation and shutdown
- control steam production
- prevent unstable operation

EPRI states that the M-MIS for the BWR RCS will usually be automatic but will provide for manual operation.

7.8 PWR Reactor Coolant System M-MIS

Section 7.8 of Chapter 10 states that the M-MIS for the PWR RCS will provide the monitoring and control necessary to

- maintain pressure control
- maintain inventory control
- remove normal and shutdown core heat

EPRI states that the M-MIS for the PWR RCS will usually be automatic but will provide for manual operation.

7.9 PWR Chemical and Volume Control System (CVCS) M-MIS

Section 7.9 of Chapter 10 states that the M-MIS for the CVCS will provide the monitoring and control necessary to

- maintain the boron concentration
- maintain RCS inventory
- control water chemistry

EPRI states that the M-MIS for the CVCS will provide indication and control as described in Section 6.5 of Chapter 3 of the passive Requirements Document. The CVCS will be a non-safety-grade system.

7.10 Process Sampling System M-MIS

Section 7.10 of Chapter 10 states that the M-MIS for the process sampling system will provide the control and monitoring necessary to sample and evaluate various fluid and gas systems. Sampling will be initiated manually and locally; however, operator aids will be provided. Interlocks will be provided to prevent sampling from affecting system availability. Repetitive monitoring will be automatic. If automatic sampling requires a system change that is also automatic, the change will be annunciated.

7.11 PWR Boron Recycle System M-MIS

Section 7.11 of Chapter 10 states that the M-MIS for the PWR boron recycle system will provide the monitoring and control necessary to process the RCS water and provide the recycled makeup water and boric acid.

7.12 BWR Reactor Water Cleanup (RWCU) System M-MIS

Section 7.12 of Chapter 10 states that the M-MIS for the BWR RWCU system will provide the monitoring and control necessary to remove reactor coolant, maintain proper inventory during operations, maintain the reactor water chemistry, and remove core decay heat during periods of reactor isolation and during cooldown and shutdown. EPRI states that RWCU operations associated with startup and shutdown will be performed in the MCR. Filter and demineralizer operations will be performed at stations outside the MCR. The RWCU system M-MIS will be capable of being monitored in the MCR. The designer will determine the operations that need to be controlled from the MCR.

EPRI states that the RWCU system will be automatically isolated if automatic monitoring detects a significant leak or if the standby liquid control system is initiated.

7.13 PWR Steam Generator System M-MIS

Section 7.13 of Chapter 10 states that the M-MIS for the PWR steam generator system will provide the monitoring and control necessary for steam production for normal turbine operation and for certain decay heat removal operations.

EPRI states that steam generator level control during startup will be automated and will include both backup and main feed pumps. Manual control will be provided. Normal level control will be automatic. Level control during planned or unplanned reactor and turbine trips will be automatic.

In its RAI dated May 17, 1991, the staff requested that the requirements of Section 7.13.2.3 address alternate means of determining inventory. The staff also discussed some of the technical difficulties experienced by automatic inventory controls. In its response dated July 22, 1991, EPRI stated that these are design choices that have to be made be the M-MIS designer. In the DSER for Chapter 10, the staff concluded that the requirement for alternate means of monitoring inventory should be included in the section, and additional guidance or specific examples in this area could be provided with the requirement and rationale. This was identified an open issue. In its letter dated September 29, 1992, EPRI stated that Section 7.13.2.3 addresses the need for automatic reactor coolant inventory control, but does not specify how this control is effected. EPRI also stated that the M-MIS designer would have to account for problems with reactor vessel level instrumentation as they arise. The staff concludes that this clarification is acceptable and considers this issue to be resolved for the passive Requirements Document. Therefore, this open issue is closed. During its review of a individual application for FDA/DC, the staff will ensure that this issue is resolved on the basis of the outcome of the ongoing efforts by the NRC and the industry to resolve the concern pertaining to the effect of entrained non-condensible gases in the

reference legs of reactor vessel level instrumentation. This issue is identified in NRC Information Notice 92-054, "Level Instrumentation Inaccuracies Caused by Rapid Depressurization," and GL 92-04, "Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in BWRs Pursuant to 10 CFR 50.54(f)."

7.14 Reactor Coolant System Leak Detection M-MIS

Section 7.14 of Chapter 10 states that the M-MIS for RCS leak detection will monitor the RCS leakage to ensure it is within acceptable limits. Because leakage from the RCS that cannot be specifically located may require plant shutdown, EPRI requires that the M-MIS reduce leakage data and present a summary of current estimated leak rates and other leakage indications that would help identify actual reactor coolant leakage. Several items are listed in the passive Requirements Document that may be used to identify the quantities and location of leakage. The staff has reviewed these items and, because they do not conflict with current regulatory guidance or requirements, concludes that they are acceptable.

EPRI states that the M-MIS for this system will be specifically designed so that the impact of plant and system transients on inventory is considered to avoid misleading leakage results.

7.15 PWR Backup Feedwater (BFW) System M-MIS

Section 7.15 of Chapter 10 states that the M-MIS for the BFW system will provide the monitoring and control necessary to remove reactor decay heat through the steam generators when the main feedwater system is not available. EPRI states that the monitoring capability provided will enable operators to manually initiate the BFW system for most events that involve the loss of main feedwater before the automatic initiation takes place.

7.16 PWR Reactor Shutdown Cooling (RSDC) System M-MIS

Section 7.16 of Chapter 10 states that the M-MIS for the PWR RSDC system will provide the monitoring and control necessary to remove core decay heat and RCS sensible heat, transfer reactor coolant to the chemical and volume control system, remove heat from the in-containment refueling water storage tank, and provide low-temperature overpressure protection. EPRI states that the RSDC system can be manually initiated and operated from either the MCR or the remote shutdown station. During normal operation, the M-MIS will not require continuous operator attention.

7.17 Conclusions

The staff concludes that the requirements in Section 7 of Chapter 10 do not conflict with current regulatory requirements and are, therefore, acceptable. However, the requirements by themselves do not provide sufficient information to make a determination that a specific design application will be acceptable. Therefore, applicants referencing the passive Requirements Document will be required to demonstrate compliance with the guidance in the SRP or provide justification for alternate means of implementing the associated regulatory requirements.

8 REACTOR PROTECTION AND SAFETY SYSTEMS M-MIS REQUIREMENTS

8.1 Purpose and Scope

Section 8.1 of Chapter 10 of the passive Requirements Document states that Section 8 provides the requirements for the M-MIS that will monitor and control reactor protection and safety systems.

EPRI specifies that the M-MIS for the systems addressed in Section 8 will be coordinated and integrated with the M-MIS for other plant systems.

8.2 General Requirements for Reactor Protection and Safety Systems Group M-MIS

8.2.1 Functions

Section 8.2.1 of Chapter 10 of the passive Requirements Document states that the M-MIS for the RPS and other safety systems will provide the monitoring and control necessary for those systems to carry out the required plant and system functions. EPRI divides the M-MIS functions into two categories: core-damage protection and core-damage mitigation. Section 1 of Chapter 10 states that the passive Requirements Document incorporates technological improvements. In its RAI dated April 10, 1990, the staff asked EPRI to provide additional background information to demonstrate that these are improvements. The staff also requested clarification of the level of safety of the I&C systems as compared with that of previous designs. By letter dated July 23, 1990, EPRI indicated that it was not evident how to determine whether the designs of the I&C systems proposed in the passive Requirements Document are safer than previous designs. However, EPRI expects that the total system and operator interface will be substantially safer than that in current plants.

EPRI states that the M-MIS will provide the monitoring and control capability to limit the reactor energy so that the core is not damaged. Since this is the basic goal of the existing RPS functions, the staff concludes that this requirement is acceptable.

Section 8.2.1 of Chapter 10 of the passive Requirements Document requires a safety-related M-MIS for pressure reduction in the reactor vessel and RCS to prevent core damage that is independent of the M-MIS that monitors and controls RCS pressure for most conditions.

EPRI requires a separate safety-related M-MIS independent of the RCS M-MIS and RCS leak detection M-MIS to maintain an adequate inventory of reactor coolant in the reactor and RCS.

Several diverse and independent systems will be available to remove decay heat. EPRI requires that the independence of the systems be maintained in the M-MIS portion by appropriate segmentation and separation. However, Section 8.2.1.4 of Chapter 10 does not address diversity of the M-MIS equipment. The staff's conclusions on the need for diversity are discussed in Section 6.1.6 of this chapter.

The M-MIS will provide the monitoring and control necessary for containment isolation, from detection of the need through initiation, accomplishment of the isolation, and confirmation of completion and maintenance of the isolation.

EPRI states that the M-MIS will provide the control and monitoring necessary to maintain the integrity of the containment boundary. This function includes containment cooling and pressure control as well as the monitoring and control of related areas such as suppression-pool level and cooling to ensure availability of the mitigation function.

EPRI states that monitoring to determine the potential for radioactivity release and the control necessary to limit the release will be provided.

8.2.2 Boundaries and Interfaces

Section 8.2.2 of Chapter 10 states that the physical boundaries that make up the safety systems are defined in Chapter 5 of the passive Requirements Document and that the M-MIS boundaries will be consistent with the physical boundaries of the plant systems.

8.2.3 Common Control and Monitoring Strategies for Reactor Protection and Safety Systems

Section 8.2.3 of Chapter 10 states that the M-MIS designer will use a consistent control and monitoring strategy. The control strategies will also be consistent with those for the non-safety M-MIS.

Section 8.2.3.1 of Chapter 10 states that the M-MIS for protection and safety systems will normally start up or actuate automatically. Manual initiation will also be provided. There will be an alarm in the MCR after initiation, and the operators will be provided information to confirm that the actuation was necessary and resulted in the expected system action. The staff concludes that this is acceptable.

The M-MIS will be designed for automatic action for the first 30 minutes after actuation. In addition, Section 4.2.2.2 of Chapter 10 states that no credit for operator action will be necessary to meet regulatory limits for at least 72 hours following the initiating events. The staff concludes that this is acceptable.

Normal shutdown functions will be accomplished manually. Testing during operation will be manually initiated but will normally be accomplished automatically during station operation. In the DSER for Chapter 10, the staff concluded that this requirement was acceptable. However, EPRI should clarify if the actuation logic will be continuously self-tested during normal operation. This was identified as an open issue. In its letter dated September 29, 1992, EPRI explained that Section 8.3.2.3 requires that the RPS provide for automatic self-test of as much of the system as is practical. EPRI stated that it expects that all, or essentially all, of the actuation logic will be self-tested automatically during normal operation. The staff concludes that this explanation clarifies the operation of continuous self-testing of actuation logic. Therefore, this open issue is closed.

Section 8.2.5 of Chapter 10 states that the designers will select the variables for automatic actuation of reactor trip that provide a balance between the need to provide diversity of actuation signals to reduce the potential for common-mode failures and the need to prevent spurious actuations. The staff concludes that this is good engineering practice; however, it will evaluate the designer's selection of these variables during its review of an application for FDA/DC to ensure that priority is given to safety.

EPRI requires that an initiated action will proceed until completion. After actuation, manual action will be required to reset the system and permit rod withdrawal.

Section 8.2.5.4 of Chapter 10 states that no single failure in the system will prevent or initiate a protection action. A second failure will not prevent a protection action; however, it may result in reactor shutdown. This second failure requirement does not apply to reactor trip or other final actuation devices, and will be limited to the non-test configuration of the system. The staff concludes that this requirement will not prevent safety systems from performing their intended function and is, therefore, acceptable.

EPRI states that the logic for the RPS and safety systems will require coincidence of two or more channels of the same variable to initiate system actuation. Coincidence of different variables will not be used. In response to a staff RAI dated July 23, 1990, EPRI provided an example of the intent of this requirement. The example demonstrated that there were more combinations of two variable inputs that would result in a trip output with different variable logic than with the same variable logic. EPRI stated that the primary consideration in the example is that a single input from two different variables (pressure and temperature) is a less valid indication of a true trip condition than two inputs from the same variable.

EPRI requires independence between manual initiation of the protection system and automatic initiation. The RPS and safety systems will operate correctly during normal operation, maintenance, postulated accidents, loss of electric power, and operation of the fire suppression system. The staff concludes that this is acceptable.

8.3 Reactor Protection System

8.3.1 System Definition

Section 8.3.1 of Chapter 10 of the passive Requirements Document indicates that the RPS will include the sensors through the final actuation breakers or relays. This is consistent with the RPS definition at existing plants. Also included is the test and diagnostic equipment needed to maintain the RPS in a state of readiness, confirm its operational status, or determine the type and location of faults. The staff agrees with the inclusion of the test and diagnostic equipment and software as part of the RPS.

EPRI states that the RPS will monitor plant parameters and determine when the reactor needs to be shut down. EPRI states that the systems with which the RPS will interface are

- neutron monitoring system
- reactor and reactor coolant systems
- main steam and main turbine-generator systems
- control rod drive system
- electric power distribution system
- HVAC

8.3.2 Performance

Section 8.3.2.1 of Chapter 10 states that the normal RPS for BWRs will provide the control signals for the control rod drive motors in addition to the primary scram control signals. Also, a backup system, which will be separate from the portion of the RPS that will provide the primary scram initiation, will provide an alternate rod injection function. This backup system will not be single-failure proof. This meets the requirements of the anticipated transient without scram (ATWS) rule (10 CFR 50.62) and is acceptable.

Section 8.3.2.2 of Chapter 10 requires that the PWR designer either provide a backup reactor trip system or demonstrate that the design of the plant has enough features so that the backup system is not warranted. The designer has the option of complying with the ATWS rule or demonstrating that the plant can withstand an ATWS event.

The RPS will provide for automatic self-testing.

8.3.3 Configuration

Section 8.3.3 of Chapter 10, as amended by EPRI's January 24, 1992, letter, specifies that the RPS will be located in a vital area, and, where practical, the segments of the system will be located in physically separate locations to make it more difficult for unauthorized personnel to disable the system without being detected. Section 5.2.4.2 of Chapter 9 also contains requirements that apply to access to these RPS segments. RPS initiation, completion, and other status information will be displayed to the MCR operators even with loss of ac power. Section 8.3.3 of Chapter 10 also requires safeguards against inadvertent or unauthorized changes in RPS setpoints, including features to ensure proper approval and recording of changes.

8.3.4 Equipment Requirements

Section 8.3.4 of Chapter 10 states that highly reliable components will be used in the RPS. The designer will identify problems associated with the existing reactor trip breakers and will establish functional and design requirements, specifications, and testing requirements to address the problems identified.

EPRI specifies that the designer will select a reactor trip device that will not degrade over the design life of the plant. In the rationale portion of this section, EPRI indicates that the primary concern with regard to these requirements is the metal-clad air circuit breakers. These circuit breakers have experienced a number of problems, including deficiencies in undervoltage trip attachments, lubricants, and manufacturing tolerances.

Section 8.3.4.3 of Chapter 10 requires the M-MIS designer to consider features in the reactor trip breakers or their enclosures that make it difficult to tamper with the breaker in a manner that would prevent its operation, provided these features are proven in service and will not prevent the breaker from tripping. The staff concludes that this is acceptable.

8.4 Passive Safety Injection (PSI) System M-MIS

Section 8.4 of Chapter 10 of the passive Requirements Document states that the M-MIS for the PSI system will provide the monitoring and control necessary to inject water into the reactor vessel to maintain sufficient inventory to prevent core damage following a loss-of-coolant accident (LOCA).

EPRI states that the PSI system will be initiated automatically, and the M-MIS will provide monitoring and control so that the operators may manually back up the initiation at both the system and component level.

EPRI states that after initiation during normal operation, the PSI system will not require the operation of automatic control systems, and manual operations may be required after 72 hours from the initiation.

The M-MIS will provide monitoring and control for manual shutdown and manual configuration changes required to return the plant functions to the usual non-safety-grade systems. Measures will be taken to prevent inadvertent manual operation. The staff concludes that the requirements of this section are acceptable.

8.5 Passive Decay Heat Removal (PDHR) System M-MIS

Section 8.5 of Chapter 10 states that the M-MIS for the PDHR system will provide the monitoring and control necessary to remove reactor decay heat when other normal systems are not available. The PDHR system will be automatically initiated and could also be manually initiated..

EPRI states that after initiation during normal operation, the PDHR system will not require the operation of automatic control systems, and manual operation may be required after 72 hours from the initiation.

The M-MIS will provide control and monitoring for manual shutdown and configuration changes required to return the decay heat removal to the usual non-safety-grade systems. Measures will be taken to prevent inadvertent manual operation. The staff concludes that the requirements of this section are acceptable.

8.6 Depressurization System M-MIS

Section 8.6 of Chapter 10 originally stated that the M-MIS for the depressurization system will provide the monitoring and control necessary to maintain the RCS pressure at a value that permits the proper operation of the PSI system, vents noncondensible gases from the pressurizer in PWRs, and opens a potential pathway so that the PDHR system can provide containment cooling in BWRs. EPRI states that the M-MIS will provide automatic and manual initiation from both the MCR and remote shutdown station. Measures will be taken to prevent inadvertent initiation. In its RAI dated May 17, 1991, the staff

asked EPRI to discuss the requirements for ensuring that the potential pathway in BWRs can be monitored and isolated. In its response dated July 22, 1991, EPRI stated that the BWR passive design has been changed so that the PDHR system will no longer be required for containment cooling. Chapter 10 will be changed to be compatible with the revision that will be made to Chapter 5 to reflect the different design approach. In the DSER for Chapter 10, the staff stated that it would confirm that this change had been made. In Revision 4, EPRI revised Sections 8.5, 8.6 and 8.10 of Chapter 10 to reflect the BWR passive design change that deleted the requirement for the PDHR system to provide containment cooling. The staff verified that the revisions address the change in the passive BWR design; therefore, this confirmatory issue is closed.

8.7 BWR Standby Liquid Control (SLC) System M-MIS

Section 8.7 of Chapter 10 states that the M-MIS for the SLC system will provide the monitoring and control for the poison injection system.

EPRI states that the M-MIS for the SLC system will be coordinated only to the minimal amount necessary to determine when to use the SLC system. The reactor water cleanup system will isolate automatically on actuation of the SLC system.

The passive Requirements Document originally stated that the SLC system would be initiated only by direct operator action to prevent inadvertent actuation. In the DSER for Chapter 10, the staff stated that this requirement was not in conformance with 10 CFR 50.62 (ATWS rule), which requires automatic initiation, and therefore was not acceptable. EPRI had not provided adequate justification in the passive Requirements Document for deviation from the rule and had not addressed any possible methods for providing highly reliable automatic initiation. This was identified as an open issue. In Revision 4, EPRI revised Sections 8.7.2 and 8.7.3 of Chapter 10, to require automatic SLC initiation. The staff concludes that the design now conforms to the ATWS rule. Therefore, this open issue is closed.

8.8 Containment Isolation M-MIS

Section 8.8 of Chapter 10 states that the M-MIS for the containment isolation system will provide the monitoring and control necessary to isolate the containment in order to minimize the release of radioactivity to the environment.

EPRI states that containment isolation will be automatically initiated and accomplished. A comprehensive operator display and appropriate controls will be provided.

8.9 Containment System M-MIS

Section 8.9 of Chapter 10 contains requirements for the monitoring and control needed to keep radioactive material inside the containment. EPRI states that the monitoring of conditions inside the containment will not require direct operator action, but information displaying the status of the containment will be provided.

8.10 PWR Passive Containment Cooling (PCC) System M-MIS

Section 8.10 of Chapter 10 states that the M-MIS for the PCC system will provide the monitoring and control necessary to remove heat from the PWR primary containment following a LOCA and from operation of the PDHR system. The PCC system will automatically actuate, and the M-MIS will provide the monitoring and control necessary for the operators to back up the initiation at the system and component level.

EPRI states that after initiation during normal operation, the PCC system will not require the operation of automatic control systems, and only monitoring to confirm operation after initiation will be required.

EPRI states that the M-MIS will provide control and monitoring for shutting down the PCC system and returning it to its original configuration manually. Measures will be taken to prevent inadvertent manual operation. The staff concludes that the requirements of this section are acceptable.

8.11 Fission Product Leakage Control (FPLC) System M-MIS

Section 8.11 of Chapter 10 states that the M-MIS for the FPLC system will provide the monitoring and control needed to limit potentially radioactive leakage to the environment. Features for automatic monitoring and automatic interlocks will be provided for the FPLC system.

8.12 Combustible Gas Control System M-MIS

Section 8.12 of Chapter 10 states that the M-MIS for the combustible gas control system will provide the monitoring and control necessary to ensure containment integrity even if combustible gas has been released during an accident.

8.13 Conclusions

The staff concludes that the requirements in Section 8 of Chapter 10 do not conflict with current regulatory requirements and are, therefore, acceptable. However, the requirements by themselves do not provide sufficient information to make a determination that a specific design application will be acceptable. Therefore, applicants referencing the passive Requirements Document will be required to demonstrate compliance with the guidance in the SRP (NUREG-0800) or provide justification for alternative means of implementing the associated regulatory requirements.

9 POWER GENERATION AND MAIN TURBINE-GENERATOR SYSTEMS M-MIS REQUIREMENTS

9.1 Purpose and Scope

Section 9.1 of Chapter 10 of the passive Requirements Document states that Section 9 provides the requirements for the M-MIS that will monitor and control the power generation and main turbine-generator systems.

EPRI specifies that the M-MIS for the systems addressed in Section 9 will be coordinated and integrated with the M-MIS for other plant systems.

9.2 General Requirements for Power Generation and Main Turbine-Generator Systems Group M-MIS

Section 9.2 of Chapter 10 states that the M-MIS for these systems will provide the monitoring and control necessary for the systems to transport steam to the main turbine and to control steam pressure and moisture content, feed and condensate system operations, and the auxiliary steam system.

Section 9.2.2 of Chapter 10 states that the physical boundaries that make up the power generation and main turbine-generator systems are defined in Chapters 2 and 13. EPRI states that the boundaries of the M-MIS will be consistent with the physical boundaries of the plant systems.

Section 9.2.3 of Chapter 10 states that these systems will use the common control strategies that the other M-MIS systems use. The operator will perform the startup functions manually. The designer will also determine the need for automatic operation during startup or shutdown. Normal operations above a low-power level will be automatic. The M-MIS will automatically reconfigure or shut down the system.

The systems will be returned to normal operation after reconfiguration or will be shut down by manual action unless the designer establishes functions that need to be automated. Any automatic return to operation will be annunciated to the operators. The operator will initiate on-line testing for these systems, but the actual testing will be automated.

9.3 Main and Extraction Steam System M-MIS

Section 9.3 of Chapter 10 states that the M-MIS for the main and extraction steam system will provide the monitoring and control needed to transport the steam to the main turbine and maintain steam quality.

EPRI states that the designer will provide for smooth transfers between automatic and manual control. The M-MIS will provide for maneuvering and load following as needed without short-term operator action. The actions of this system will be compatible with those of the reactor and safety systems. The control strategy will be compatible with on-line testing.

9.4 Main Turbine System M-MIS

Section 9.4 of Chapter 10 states that the M-MIS for the main turbine will provide the monitoring and control for the main functions of the turbine.

9.5 Main Generator System M-MIS

Section 9.5 of Chapter 10 states that the M-MIS for the main generator system will provide the monitoring and control for the main generator.

9.6 Feedwater and Condensate System M-MIS

Section 9.6 of Chapter 10 states that the M-MIS for the feedwater and condensate system will provide the monitoring and control needed to transport high-quality feedwater from the condenser hotwell to valves that control the introduction of feedwater into the BWR reactor vessel or a PWR steam generator.

EPRI requires that operation at low-power levels not require excessive operator attention. Features will be provided that will ensure that transfer between manual and automatic control will be smooth. The feedwater and condensate system will be automatically adjusted during normal power changes without short-term operator action.

9.7 Chemical Addition System M-MIS

Section 9.7 of Chapter 10 states that the M-MIS for the chemical addition system will provide the monitoring and control needed to maintain the condensate, feedwater, and off-gas chemistry within required limits.

The designer will determine the operations that will be controlled from the MCR instead of doing it locally. Automatic operation of the chemical addition system will not cause a shutdown of other plant systems.

9.8 Condensate Makeup and Purification System M-MIS

Section 9.8 of Chapter 10 states that the M-MIS for the condensate makeup and purification system will provide the monitoring and control needed for makeup supply and chemistry.

Normal operation will be automatic. Automatic actions of this system, including component failures, will not cause a shutdown of other plant systems unless the shutdown is necessary to prevent hazards to personnel or damage to other equipment.

9.9 Auxiliary Steam System M-MIS

Section 9.9 of Chapter 10 states that the M-MIS for the auxiliary steam system will provide the monitoring and control necessary to ensure an adequate supply of auxiliary (low-pressure) steam is available to the plant systems that utilize auxiliary steam to perform their functions.

EPRI expects that most or all of these functions will be performed outside the MCR. The designer will stipulate any functions that are essential enough to be included in the MCR. Automatic actions of this system, including component failures, will not cause a shutdown of other plant systems unless the shutdown is necessary to prevent hazards to personnel or damage to other equipment.

9.10 Conclusions

The staff concludes that the requirements in Section 9 of Chapter 10 do not conflict with current regulatory requirements and are, therefore, acceptable. However, the requirements by themselves do not provide sufficient information to make a determination that a specific design application will be acceptable. Therefore, applicants referencing the passive Requirements Document will be required to demonstrate compliance with the guidance in the SRP (NUREG-0800) or provide justification for alternative means of implementing the associated regulatory requirements.

10 AUXILIARY AND PLANT SUPPORT SERVICES SYSTEMS M-MIS REQUIREMENTS

10.1 Purpose and Scope

Section 10.1 of Chapter 10 of the passive Requirements Document states that Section 10 provides the general requirements for the M-MIS that will monitor and control the auxiliary and plant support services systems. The auxiliary systems covered by this section include the plant cooling water system; the compressed air and gas systems; the HVAC system; and the electric power systems. The plant support services systems include the fuel handling systems, the fire protection system, the environmental monitoring system, the site security system, and the radioactive waste processing systems.

10.2 General Requirements for Auxiliary and Plant Support Services Systems Groups M-MIS

10.2.1 Functions

The M-MIS will provide the monitoring and control for the systems listed in Section 10.1. Because the function of each M-MIS for these systems is self-explanatory, the staff has not addressed them individually. However, the following discussion addresses the staff's concerns regarding certain requirements proposed by EPRI for the M-MIS for specific systems.

Section 10.2.1.8 of Chapter 10 lists the security-related functions of the plant support service group M-MIS, including preventing unauthorized access into vital or protected areas, detecting attempts to gain unauthorized access or introduce unauthorized materials into such areas, facilitating authorized activities and conditions within protected and vital areas, and providing authorized access and detection of unauthorized penetration of protected or vital area boundaries. Additional security-related functions are implicitly included by reference to Chapter 9, Section 5, of the passive Requirements Document.

In Revision 1 of Appendix B to Chapter 1 of the passive Requirements Document, EPRI has committed to meet the guidelines of SRP Sections 12.3 and 12.4, "Radiation Protection Design Features," which provide detailed guidance on an acceptable area radiation monitoring system. Because EPRI has committed to meet the guidelines of SRP Sections 12.3 and 12.4 and because it is outside the scope of the Requirements Document to provide detailed information on the area monitoring equipment to be used at the ALWR, the staff concludes that this issue is resolved for the passive Requirements Document. It will be the responsibility of each applicant for design certification to provide the detailed information on the area monitoring system that is called for in Sections 12.3 and 12.4 of the SRP. Therefore, the staff will evaluate this matter during its review of an individual application for FDA/DC.

In addition, Item II.F.1-3 of NUREG-0737 states that licensees must be able to detect and measure the radiation levels in the containment during and following an accident. In Revision 1 of Appendix B to Chapter 1, EPRI committed to comply with 10 CFR 50.34(f)(2)(xvii), which addresses Item II.F.1-3 of NUREG-0737 on accident monitoring instrumentation. Because the description of the specific design features necessary to comply with NUREG-0737 is the

responsibility of each applicant for design certification, the staff will evaluate this matter during its review of an individual application for FDA/DC.

Sections 12.3 and 12.4 of the SRP describe the criteria for an acceptable airborne radioactivity monitoring system. These criteria address monitor location, placement, calibration frequency, and alarm and readout for airborne radioactivity monitors. In addition, these SRP sections include emergency power requirements and setpoint levels for these monitors. In Revision 1 of Appendix B to Chapter 1, EPRI committed to meet the guidelines of SRP Sections 12.3 and 12.4, which provide detailed guidance on an acceptable airborne radiation monitoring system. Because it is outside the scope of the Requirements Document to provide detailed information on the airborne radioactivity monitoring system, this information must be provided by each applicant for design certification. The staff will evaluate this matter during its review of an individual application for FDA/DC.

10.2.2 Boundaries and Interfaces

Section 10.2.2 of Chapter 10 states that the physical boundaries that will make up the auxiliary and plant support services systems are defined in Chapters 7, 8, 9, 11, and 12 of the passive Requirements Document. EPRI states that the boundaries of the M-MIS for these systems will be consistent with those physical boundaries of the plant systems.

10.2.3 Control and Monitoring Strategies for Auxiliary and Support Systems

Section 10.2.3 of Chapter 10 states that a consistent control strategy will be used. These systems will primarily be operated from outside the MCR. Monitoring and control will be provided for normal startup and shutdown. Some tasks may be automated. Normal operations for these systems will be performed automatically.

The M-MIS will provide automatic reconfiguration when needed for investment protection or safety. In general, on-line testing will be operator initiated, but the testing will be automated.

Security Concerns

The M-MIS used to control the security functions of the auxiliary and plant support services systems needs to be attended or staffed on a continuous basis, although this would be done at the security alarm stations rather than at the MCR. In the DSER for Chapter 10, the staff concluded that EPRI should modify Section 10.2.3.2 of Chapter 10 in the manner discussed in EPRI's January 28, 1992, letter to clarify that this requirement is intended to relieve operators or security personnel from spending their time attending to the operation of the security M-MIS, rather than using the security M-MIS to carry out their security functions. The staff stated that it would confirm that this change had been made in future revisions of the passive Requirements Document. The staff has confirmed that this change has been made to Chapter 10; therefore, this confirmatory issue is closed.

10.2.4 Integration and Coordination

Section 10.2.4 of Chapter 10 states that the M-MIS for the systems addressed in Section 10 will be integrated and coordinated with the M-MIS for other systems.

10.2.5 Independence and Redundancy Requirements

Section 10.2.5 of Chapter 10 states that the M-MIS will be independent and redundant as defined for the particular system. Failures in the M-MIS for the systems addressed in Section 10 will not cause plant unavailability or challenge the reactor protection or engineered safety feature actuation systems.

Specific requirements for security M-MIS independence are contained in 10 CFR 73.55(e)(1), which requires that the security central alarm station not contain any other activities that could interfere with its security functions. Section 5.2.13.1 of Chapter 9 of the passive Requirements Document states that all security central processing units will be dedicated to security functions only. However, the individual system chapter for the security system does not contain requirements for independence for data transmission between protected area perimeter intrusion detection sensors and the central and secondary alarm stations (CAS/SAS). As discussed in Sections 5.2.1 and 5.2.5 of this chapter, independent data transmission of these alarms and associated CCTV signals may be necessary to ensure a capability to meet the alarm assessment requirements of 10 CFR 73.55(h)(4)(i) and (ii). Although this potential need for independence is not covered explicitly by Section 10.2.5 of Chapter 10 of the passive Requirements Document, it does not prohibit this independence and, thus, does not conflict with NRC requirements.

10.2.6 Fire Protection and Security

Section 10.2.6 of Chapter 10 states that the M-MIS will enhance the plant capability for fire protection and security by providing alternate controls and monitoring so that fires and fire-suppression actuation would not require plant shutdown because of the unavailability of M-MIS equipment. This section also states that the M-MIS design will consider features to prevent a single security breach from disabling or challenging a complete safety system or function.

10.3 Conclusions

The staff concludes that the requirements in Section 10 of Chapter 10 do not conflict with current regulatory requirements and are, therefore, acceptable. However, the requirements by themselves do not provide sufficient information to make a determination that a specific design application will be acceptable. Therefore, applicants referencing the passive Requirements Document will be required to demonstrate compliance with the guidance in the SRP or provide justification for alternative means of implementing the associated regulatory requirements.

11 CONCLUSION

On the basis of its review, the staff concludes that the EPRI requirements established in Chapter 10 of the passive Requirements Document for the design of the M-MIS do not conflict with current regulatory guidelines and are acceptable. However, by themselves they do not provide sufficient information for the staff to determine if the design and arrangement of plant-specific M-MIS will be adequate. Applicants referencing the passive Requirements Document will be required to demonstrate compliance with the guidance in the SRP or provide justification for alternate means of implementing the associated regulatory requirements.

Therefore, the staff concludes that Chapter 10 specifies requirements that, if properly translated into a design in accordance with the NRC regulations in force at the time the design is submitted, should result in a nuclear plant whose M-MIS will perform as designed and have all the attributes required to ensure that there is no undue risk to the health and safety of the public or to the environment.

APPENDIX A
DEFINITIONS OF TERMS AND ACRONYMS

Appendix A of Chapter 10 of the passive Requirements Document contains definitions of terms and acronyms. The staff has provided a consolidated list of acronyms in Volume 1 of this report.

APPENDIX B
HUMAN FACTORS ASSESSMENT OF THE EPRI ALWR PASSIVE REQUIREMENTS DOCUMENT

1 INTRODUCTION

This appendix gives the results of the staff's review of the human factors aspects of the passive Requirements Document (Volume III of the EPRI Requirements Document). The scope of the staff's review is based on Chapter 10 of the passive Requirements Document through Revision 4.

The staff's review of the passive Requirements Document was based on the regulatory requirements established in 10 CFR 50.34(f)(2)(iii), 10 CFR 50.34(g) and 10 CFR 52.47 and the guidance in Sections 13 and 18 of NUREG-0800, "Standard Review Plan [SRP] for Nuclear Power Plants" and NUREG-0700, "Guidelines for Control Room Design Reviews."

2 STAFF REVIEW OF HUMAN FACTORS IN ADVANCED PLANTS

Current regulatory human factors guidance is contained in Sections 13 and 18 of the SRP. Section 18, "Human Factors Engineering," provides guidance for the review of existing control rooms, remote shutdown stations, and safety parameter display systems. Its focus is to ensure that personnel-equipment interfaces conform to accepted human factors practices (i.e., practices accepted by human factors professionals). Section 13, "Conduct of Operations," contains review guidance for issues related to the qualification of operators; the training of plant staff; and normal, abnormal, and emergency operating procedures. NUREG-0700, which is referenced in SRP Section 18, contains guidance on the use of systems and operations design analysis techniques for existing and new (Appendix B) control room designs. The staff used these guidance documents during its review of Chapter 10 of the passive Requirements Document.

Sections 13 and 18 of the SRP were developed to provide evaluation criteria for limited backfits and modifications of existing nuclear power plants. Since the SRP does not contain guidance for evaluating new control rooms, the staff developed new positions based on accepted human factors principles and practices for new plant designs using new technology such as touch screens and cathode ray tubes. These positions were applied in the staff's review.

One area of consideration has been the coordination and execution of the programmatic elements of a human factors program. In the past, a limited backfit approach was used to evaluate existing control room designs. For new control room designs, the staff has taken the position that a human factors effort should be conducted within the context of a larger full-scope plant systems engineering effort. The human factors program should be integrated into all aspects of an advanced reactor plant design and be an active contributor to achieving the design goals of increased plant safety and reliability.

A reduction in the probability of operator error, demonstrated through an effective human factors program, is an important element in an acceptable advanced passive reactor design.

Advanced reactor M-MIS should be designed only after a systems analysis has been performed. The human factors portion of a systems analysis should include a function analysis, a function allocation, and a V&V of the function allocation. The functions defined by the function analysis provide the basis for subsequent task analyses. The goal of using a systematic approach to the definition of human factors requirements is to ensure that the information and control requirements needed for control station procedures and training are sufficient and support the operators in accomplishing required activities safely, effectively, and reliably. Only through a systematic and properly organized approach can an advanced reactor designer document and demonstrate that human performance has been appropriately considered in the design.

Item 2 of Section 18.1 of the SRP states that in new advanced reactor control stations, advanced display and control technologies should be used and that the station should support operator functions and tasks and aid decisionmaking under all plant operational modes and conditions. The plant designer must evaluate the concepts and technologies embodied in the use of computerized workstations, digital technology - such as electronic displays and "soft" controls and synthesized data - to ensure that they are technically feasible, that they are within the state of the art, that they provide the capabilities addressed in Item 2 of Section 18.1, and that they conform to accepted human factors principles. As with the programmatic elements, the reactor designer should document the technical elements of the man-machine interfaces and demonstrate that human performance has been appropriately accounted for in the design.

Sections 1.4.1 and B1 of NUREG-0700 advocate that a human factors program plan be developed as the first step of the control room design review. Similarly, the staff has taken the position that advanced reactor designers should include a human factors program plan as part of the plant design effort. The program should cover all aspects of human factors that contribute to plant safety and reliability, and address at a minimum (1) man-machine interfaces for personnel interactions and interventions at the function, system, and component levels; (2) staffing and qualification requirements for personnel; (3) development of procedures and training for normal, abnormal, and emergency modes of plant operation; (4) maintenance of plant equipment; and (5) development of training and procedures for maintenance tasks. In response to a RAI, EPRI, by letter dated November 15, 1991, stated that the operating philosophy for passive plant designs is the same as that for evolutionary plant designs, since the monitoring functions, decisions, and actions required by the operator continue to be based on the status of the plant, except for the detail and timing of actions performed by the operators. EPRI stated that the passive Requirements Document reflects these differences for both safety and non-safety systems. The staff finds the EPRI position on the similarities in operating philosophies for both the evolutionary and passive designs acceptable. However, the staff will evaluate the operating philosophy issue during its review of an individual application for FDA/DC and during licensing reviews.

This policy issue is discussed further in SECY-93-087, and as stated in the SECY paper and throughout this appendix, the staff has taken the position that sufficient man-in-the-loop testing must be performed to verify and validate passive control room designs, including the operating philosophy adopted for

that design. In addition, a fully functional integrated control room prototype is likely to be necessary for passive plant control room designs to demonstrate that functions and tasks are properly integrated into the man-machine interface design.

Consistent with Section 2.1 of Appendix A to Section 18 of the SRP and Sections 2.3 and B3 of NUREG-0700, the staff considers it essential that a human factors program include full participation of qualified human factors specialists during all phases of the design process.

The human factors program should also include the necessary human factors tests and evaluations to verify that anticipated improvements in human performance have been achieved. This approach conforms to the guidance in Sections 3.7, 3.8, and B4.3.2.6 of NUREG-0700.

The staff has taken the position that the ALWR passive plant designer's human factors program should include the development of a fully operational control room prototype. ALWR control rooms are expected to incorporate state-of-the-art technologies and technologies that have been proven in other industries. However, the synergistic effects of these new combinations of equipment must be thoroughly evaluated. A fully operational control room prototype should be used to accomplish the human factors tests, evaluations, and validations necessary to ensure that the ALWR control room is (1) consistent with the plant operating philosophy, (2) supports operator activities under all plant operational modes and conditions, (3) provides for the development of standardized training and procedures, and (4) provides the necessary assurance that an appropriate reduction in operator error has been achieved over current designs. Similar guidance is promulgated in Sections 3.8 and B4.4 of NUREG-0700. In response to a RAI, EPRI, by letter dated November 15, 1991, described a staged approach to the use of mockups, simulators, and prototypes. Although not directed specifically at meeting licensing milestones, the staff finds the approach acceptable. However, it will evaluate the issue of mockups, prototypes, and simulators during its review of an individual application for FDA/DC and during licensing reviews.

3 EVALUATION

The staff's human factors evaluation of the passive Requirements Document is contained in the following sections. Unless otherwise stated, sections referenced in the evaluation are from Chapter 10 of the passive Requirements Document.

3.1 Composition, Organizational Structure, and Scheduling of the Human Factors Function

3.1.1 Composition of the Human Factors Function

On the basis of the guidance in Section 2.1 of Section 18.1 of the SRP, Sections 2.3 and B3 of NUREG-0700, and accepted human factors principles and practices, the designer should identify and describe the organizational structure, personnel, and methods and criteria to be used to integrate human factors into the design process. At a minimum, the following elements should be included as part of the human factors organizational structure:

- reporting responsibilities and authority of the human factors staff qualifications and number of human factors personnel and job descriptions for each position

- role of human factors in a multidisciplinary design team

Sections 3.1.2.2, 3.1.2.3, 3.1.4.1 and 4.1.3.1 of Chapter 10 of the passive Requirements Document specify that the M-MIS designer will organize and direct design teams for particular major interfaces, and establish the MCR design team, the independent review team, the qualifications of the independent review team, and the makeup of the control station review team.

Generally, each of the four sections referenced above specifies the makeup of the team(s) by engineering and design disciplines and specialties, as well as inclusion of human factors specialist(s) and plant owner staff representatives. However, the passive Requirements Document does not contain a requirement that computer and hardware specialists be included on the design and review teams. Given the amount of hardware and software that will be used in an ALWR, as currently envisioned by the industry and required by Section 6 of Chapter 10, not requiring a computer specialist to be on the design or review teams could result in this area not receiving adequate consideration in the development of the control room design. By letter dated March 19, 1992, EPRI revised the team composition to specifically include computer hardware and software specialists. The staff finds the change acceptable and closes out the issue regarding the inclusion of computer specialists on the design and review teams.

3.1.2 Organizational Structure of the Human Factors Function

Section 3.1.2.4 of Chapter 10 requires that the M-MIS designer prepare a comprehensive plan for developing and implementing the M-MIS design that includes elements for scheduling, configuration control, design, development and testing tools, design team interfacing and interactions, and participation of utility representatives. In response to an August 30, 1990, RAI, EPRI, by letter dated December 6, 1990, committed to include in Section 3.4.1 of Volume I of the Requirements Document, "ALWR Top Level Design Process Requirements," a requirement that the plant designer have a documented plan for developing and implementing the ALWR design.

In the DSER for Chapter 10, the staff determined that EPRI's response was inadequate and stated that the plan should clearly identify the responsibilities and authorities for all aspects of the design process and specifically identify the features that are incorporated to ensure a systematic consideration of human factors, including (1) the reporting responsibilities and authority of the human factors staff, (2) the qualification and number of human factors personnel, (3) job descriptions for each position, (4) the role of human factors in a multidisciplinary team, and (5) the plans for testing to substantiate the human factors design features of the plant. This was identified as an open issue in the DSER for Chapter 10. In a letter dated March 2, 1993, EPRI revised Section 11.8.5 of Chapter 1 to include these requirements and referenced additional EPRI guidance documents including EPRI NP-3659, "Human Factors Guide for Nuclear Power Plant Control Room Development," and appendix A, "Human Factors Maintainability Program Plan (HFMPP) Requirements," of EPRI NP-4350, "Human Engineering Design Guidelines for

Maintainability." EPRI considers that these provide ALWR vendors with the detailed guidance necessary for a design. Applicants for FDA/DC must demonstrate that the organization and structure of their human factors function meet these requirements in accordance with the staff guidance titled "HFE Program Review Model and Acceptance Criteria for Evolutionary Reactors." Therefore, this DSER open issue is closed.

3.1.3 Scheduling of Human Factors Studies and Evaluations

Section 3.1 of Chapter 10 requires that the M-MIS design be established by a defined process that begins at the same time as the rest of the plant design process and that meets the requirements of Section 3 of Chapter 10 and the overall design process requirements of Chapter 1 of the passive Requirements Documents. In response to an April 17, 1991, RAI, EPRI by letter dated June 13, 1991, stated that although human factors research studies were important in setting the general direction of the design, it anticipated that the studies performed as part of the V&V process would define the final control room configuration.

The requirements of 10 CFR 50.34(f)(2)(iii) state that a control room design should reflect state-of-the-art human factors principles before committing to fabrication of control room panels and layouts. Item 2 of Section 18.1 of the SRP states that control room designs shall reflect human factors principles by specifying the design concept selected and providing supporting bases and criteria.

In the DSER for Chapter 10 of the passive Requirements Document, the staff described its position with respect to the scheduling of the major portion of human factors studies in support of the ALWR designs. The staff concluded that a major portion of the human factors studies should be completed before the start of the control room design process and that the results of the human factors studies should be available to the M-MIS designer during the early phases of the design process when critical decisions are being made concerning the conceptual framework, baseline design of the control room, the design of the control boards, and the depth and breadth of hardware and software systems.

Incorporating man-machine interfaces into the control room using a V&V approach places human factors in a retrofit mode from the very beginning of the design process. Such an approach tends to focus the human factors effort on specific issues rather than on the control room as a whole. The studies and evaluations performed before the start of the actual design process contribute valuable insights concerning the human factors aspects of the individual man-machine interfaces considered for the control room, as well as the synergistic effects created by various combinations of the interfaces within the control room.

In response to the DSER for passive plants and as a result of the November 3, 1992, meeting between the staff and EPRI, Section 2.2.10 of Chapter 10 was revised to state that the specific design features described in the passive Requirements Document are to be used as a starting point for the design. As part of the M-MIS design process, the human factors studies and human performance evaluations are to be used as the basis for the actual design features selected. Most importantly, the requirements on the design process will take

precedence over the specific features identified as the starting point. Modifications to the individual design features, described in the passive Requirements Document, as a result of the implementation of a specific M-MIS design process, are acceptable. The staff has reviewed the revisions to the passive Requirements Document with respect to the scheduling of human factors studies in support of the M-MIS design and finds that EPRI's response addresses its concerns and is, therefore, acceptable. However, the applicant for FDA/DC will be responsible for documenting the technical basis for all design feature decisions resulting from the implementation of its M-MIS design process. As part of its review of individual applications for FDA/DC, the staff will examine the technical basis for design features. Therefore, this open issue is closed.

3.1.4 Interfaces With Other Groups

In Section 3.1 of Chapter 10, EPRI states that experience has shown that conventional design methods cannot be expected to provide good interfaces between the operators and the plant and that even if all the design requirements are identified, it is unrealistic to expect them to be met in a simple and practical manner unless the design process is systematic and consistent. The staff has identified two important problems in implementing the EPRI requirements as part of a design process.

The first problem is the lack of guidance provided for the M-MIS designer to ensure that past problems related to the implementation of large numbers of design requirements can be identified and successfully overcome. The second problem is that the individual requirements do not provide the M-MIS designer with the necessary overall guidance to establish appropriate programmatic and technical vehicles to facilitate and achieve the goal of improving design methods and interfaces between the operator and the plant. In response to an April 17, 1991, RAI about these two problems, EPRI stated in a letter dated June 13, 1991, that the specific programmatic methods used in the design process are the responsibility of the M-MIS designer. The staff concluded that EPRI's response did not adequately address these two staff concerns; therefore, they were identified as open issues in the DSER. During a November 3, 1992, meeting between the staff and EPRI, the deficiencies in the passive Requirements Document regarding the necessary guidance for the M-MIS designer to (1) overcome past problems and (2) establish appropriate methods of interface and exchange were discussed further. As a result, EPRI revised Section 3.1.2.4 of Chapter 10 to state that appropriate administrative systems will be in place for ensuring that human factors issues identified during the design process are adequately communicated, tracked, and resolved systematically. Included in this revision is the requirement to define how the design team will communicate and coordinate human factors issues and specifically describe the requirements on the technical content of those communications. The staff has reviewed the revisions with respect to the guidance to establish appropriate methods for interface and information exchange to overcome past problems and finds EPRI's response addresses the staff's concerns and is, therefore, acceptable. Therefore, both of these open issues are closed.

3.2 Systems Analysis

Consistent with Sections 3.4 and B4 of NUREG-0700 and accepted human factors principles and practices, the plant designer should perform a human factors

analysis, including a function and task analysis, that includes an analysis of plant and system functions, system information flow, and information required for the effective and timely conduct of tasks by operations and maintenance personnel. The systems analysis should include, at a minimum, the following elements:

- Interactions required between and among systems to meet plant and personnel performance objectives should be identified.
- Functions and tasks should be defined and allocated to personnel or equipment.
- System processes should be identified and allocated to functional, system, or component levels of control.
- Personnel workload should be compared with equipment design constraints and limitations to ensure plant performance objectives are met.
- Task analyses should be performed including the identification of critical tasks and their analysis for equipment design characteristics that exceed human control capabilities or approach limitations.
- Human factors should be integrated into the design of hardware and software.

The staff's evaluation of the incorporation of these elements in the passive Requirements Document follows.

3.2.1 Identification of Requirements

The "Introduction" section of Chapter 10 states that the EPRI requirements incorporate technological improvements and human factors considerations to resolve problems experienced in today's operating LWRs. In its RAI dated April 17, 1991, the staff asked how EPRI had identified the problems and how it had generated corresponding requirements. By letter dated June 13, 1991, EPRI responded that a great majority of the problems were common knowledge and the requirements were developed through an extensive process of synthesis, review, and consensus building. EPRI also stated that the great majority of the human factors requirements in Chapter 10 were not the consequence of human factors research, but were specific statements of guidance as applied to the ALWR design. It stated that it was impractical to perform research studies to support each and every requirement imposed at the beginning of the design process. In the case of the man-machine interface, and especially in the MCR, the goal of EPRI is to achieve a major, not merely an incremental, improvement. EPRI does not believe this can be done if the design is constrained to those features for which all the fundamental research and verification have been done at the beginning of the design process.

Item 2 of Section 18.1 of the SRP requires applicants starting the control room design process to reflect human factors principles in the design concept and identify the supporting bases and criteria. In the DSER for Chapter 10, the staff stated that its position on advanced reactors and human factors requirements was that potential human factors design requirements levied on a design should conform to established human factors guidance (e.g.,

professional standards and military standards), principles, practices, or applicable studies. If no principles, practices, or studies were available, the plant designer should perform the human factors studies or evaluations necessary to establish suitability of a requirement to be included as part of the design basis. The passive Requirements Document did not fully identify the human factors criteria, guidance, principles, or practices that were used as the supporting bases for many of the man-machine interface requirements. This was identified as one open issue. Section 2.2.10 of Chapter 10 has been revised to state that the specific design features described in the passive Requirements Document are to be used simply as a starting point for the design. As part of the M-MIS design process, the human factors studies and human performance evaluations are to be used as the basis for the actual design features selected. Most importantly, the requirements on the design process will take precedence over the specific features identified as the starting point. The staff has reviewed the revisions to the passive Requirements Document with respect to human factors requirements on the M-MIS design and finds the revision provided by EPRI addresses its concerns and is, therefore, acceptable. However, the applicant for FDA/DC will be responsible for documenting the technical basis for all design feature decisions resulting from the implementation of its M-MIS design process. As part of its review of individual applications for FDA/DC, the staff will examine the technical basis for design features. Therefore, this open issue is closed.

3.2.2 Basis of Human Factors Requirements

Item 2 of Section 18.1 of the SRP requires applicants to specify the design concept selected and provide supporting design bases and criteria. In the DSER for Chapter 10, the staff stated that its position was that human factors requirements levied on a control room design should identify the basis or human factors principles addressed by the requirements. The bases should reflect some aspect of a human factors principle, practice, or study that, through its application, will contribute to improved operator performance or reduce the risk of human error. The plant designer should review the basis for the human factors requirement before making design tradeoffs that involve human performance issues. The staff stated that the basis for many of the human factors requirements cited in Chapter 10 was not indicated and identified this as an open issue.

A result of the November 3, 1992, meeting between the staff and EPRI, the passive Requirements Document was revised to state that the human factors requirements described in the passive Requirements Document are to be used simply as a starting point for the design. As part of the M-MIS design process, the human factors studies and human performance evaluations are to be used as the basis for the actual human factors requirements. The staff has reviewed the revisions to the passive Requirements Document with respect to human factors requirements on the M-MIS design and finds that EPRI's response addresses the above concern and is, therefore, acceptable. Therefore, this open issue is closed. However, the applicant for FDA/DC will be responsible for documenting the technical basis for all human factors requirements resulting from the implementation of its M-MIS design process. As part of its review of individual applications for FDA/DC, the staff will examine the technical basis for these requirements.

3.2.3 Method for Establishing Human Factors Requirements

Chapter 10 of the passive Requirements Document contains (1) a description of a process for establishing human factors design requirements and (2) design requirements that are unsupported by human factors guidance, principles, or practices. In the DSER for Chapter 10, the staff concluded that the analyses performed during the design process should be the vehicles to generate the human factors requirements, which determine the man-machine interfaces incorporated into the final design. To impose human factors requirements (e.g., multiple, compact, redundant workstations and the use of electronic procedures) on the ALWR design before starting the design process negates much of the purpose and benefit of performing the systems and human factors analyses. The staff stated that EPRI should provide additional information on the method to establish effective human factors requirements on the design and identified this as an open issue.

The issue was discussed further at the November 3, 1992, meeting between the staff and EPRI and, as a result, Section 2.2.10 of Chapter 10 was revised to state that as part of the M-MIS design process, the human factors studies and human performance evaluations are to be used as the basis for the actual human factors requirements levied on the design. Most importantly, the passive Requirements Document states that the design process (e.g., human factors analyses) will take precedence over the specific features identified as the starting point. The results of these human factors analyses will be the vehicle for generating the human factors requirements on the design. The staff has reviewed the revisions to the passive Requirements Document with respect to establishing human factors requirements on the M-MIS design and finds that EPRI's response addresses its concerns and is, therefore, acceptable. Therefore, this open issue is closed. However, the applicant for FDA/DC will be responsible for documenting the human factors requirements resulting from the implementation of its M-MIS design process. As part of its review of individual applications for FDA/DC, the staff will examine the technical basis for these human factors requirements.

3.2.4 Identification of Interactions Between and Among Systems

Section 3.1.3.3 of Chapter 10 requires that the M-MIS design process result in the explicit identification of the functions of the M-MIS needed to control the plant systems and the overall operation of the plant. Section 7.2.4 specifies that the M-MIS designs for the reactor and RCS will be integrated and coordinated so that the overall plant performance and functional requirements, as well as those for the individual systems, are met. Plant systems requirements are located in other chapters of the passive Requirements Document. The specific methods and logic used to integrate the operation of the various systems are to be selected by the M-MIS designer.

In its RAI dated August 30, 1990, the staff questioned the human factors analysis methods, techniques, and documentation requirements proposed by EPRI to implement the human factors policy and top-tier requirements of the passive Requirements Document. By letter dated December 6, 1990, EPRI stated that it is the responsibility of the plant designer to establish the human factors analysis methods, techniques, and documentation. The passive Requirements Document provides policy and top-tier requirements for human factors and other requirements that affect human factors. EPRI also noted that specific

requirements on human factors analysis methods, techniques, and documentation are included in other chapters of the passive Requirements Document as appropriate, and in some cases, other EPRI guidance documents are referenced to provide further details.

In the DSER for Chapter 10, the staff concluded that the passive Requirements Document did not contain (1) specifications for a systems analysis that would be part of a human factors program for the ALWR design and (2) specific guidance on the organization of plant information to identify systems and their functional interactions in order to meet plant and personnel performance objectives. The staff stated that the passive Requirements Document should provide guidance for performing a systems analysis and organizing plant information and identified these two items as open issues.

At the November 3, 1992, meeting between the staff and EPRI, the staff requested that EPRI revise the passive Requirements Document to provide guidance for performing a systems analysis and organizing plant information. As a result, Section 3.1.2.4 of Chapter 10 was revised to emphasize that a systems analysis approach is to be used in the M-MIS design and requires that the overall structure of the approach to systems analysis and implementation of systems analysis be described. A reference to Appendix B of NUREG-0700, "Systems/Operation Design Analysis Techniques," was included in the passive Requirements Document in support of this requirement. The staff has reviewed the revisions with respect to establishing a systems analysis approach as part of the M-MIS design process and finds that EPRI's response addresses its concerns and is acceptable. Therefore, these two open issues are closed.

3.2.5 Definition of Functions and Tasks and Their Allocation to Personnel and Equipment

Sections 3.1.3.3, 3.1.3.3.1, 3.1.3.3.3, 3.4.3, 3.4.4, and 3.7.7.1 of Chapter 10 specify that the M-MIS design process is to result in (1) the identification of the M-MIS operational and maintenance functions and the individual tasks necessary to perform the functions, (2) the use of the identified functions and tasks, (3) the selection of automatic or manual control, and (4) the selection of remote or local control. The staff concludes that these requirements are consistent with accepted human factors principles and practices and the guidance in Sections 3.4, B4.1, and B4.2 of NUREG-0700 and are, therefore, acceptable.

Allocation of Functions and Tasks to Hardware and Software

In its RAI dated April 17, 1991, the staff asked EPRI to identify the primary drivers that led to the particular combination of features for the MCR described in Section 2.3 of Volume I. Section 2.3 describes the man-machine interface as using modern digital technology and an advanced control concept in which integrated displays, alarms, procedures, and controls are available to the operators at compact workstations. By letter dated June 13, 1991, EPRI responded that the primary drivers of the features identified were (1) distance between controls and displays and traffic in the control room, (2) availability of information where needed, (3) consolidation of controls and displays, (4) relationships of controls to displays, (5) consolidation and integration of information from multiple sensors, (6) consolidation of panel and computer alarms, and (7) presentation of procedures.

Sections B4.2.2 and B4.3.1 of NUREG-0700 state that system processing capabilities and failure modes should be evaluated as part of the function and task allocation process. Before the function and task analysis is completed, the computer processing capabilities are defined, and an analysis of software and hardware failure modes is performed, it is premature to establish requirements for the operators' workstations. In the DSER for Chapter 10, the staff stated that its position was that the capabilities and limitations of hardware, software, and the human component must be understood before committing critical information, controls, and displays required by the operator for the safe operation and shutdown of the plant to one particular configuration. Before the completion of the above analyses, the staff considered it premature to establish detailed requirements for operator workstations and, therefore, this was identified as an open issue.

The issue was discussed further at the November 3, 1992, meeting between the staff and EPRI and, as a result, Section 2.2.10 of Chapter 10 was revised to state that the specific design features (e.g., operator workstation design features) described in the passive Requirements Document are to be used simply as a starting point for the design. As part of the M-MIS design process, the human factors studies and human performance evaluations are to be used as the basis for the actual operator workstation design features selected. Most importantly, the requirements on the design process will take precedence over the specific features identified as the starting point. The staff has reviewed the revisions with respect to human factors requirements on the M-MIS design of the operator workstation and finds that EPRI's response addresses its concern and is acceptable. Therefore, this open issue is closed. However, the applicant for FDA/DC will be responsible for documenting the technical basis for all operator workstation design feature decisions resulting from the implementation of its M-MIS design process. As part of its review of individual applications for FDA/DC, the staff will examine the technical basis for operator workstation design features.

3.2.6 Identification of System Processes and Their Allocation to Functional, System, or Component Level of Control

Section 3.1.1 of Chapter 10 specifies that the detailed M-MIS design development process will ensure that the functional requirements of plant systems and other design requirements are met. Section 3.1.1.1 requires that the M-MIS design process be structured to emphasize the functional division of the plant such as reactivity control, reactor coolant pressure control, reactor coolant inventory and chemistry control, and reactor core heat removal for the overall control of the reactor. The staff concludes that these requirements are consistent with accepted human factors principles and practices and are, therefore, acceptable.

3.2.7 Comparison of Personnel Workload Tasks to Equipment Design Constraints

Section 2.1.1 of Chapter 10 requires that the M-MIS design take full advantage of operator capabilities but not challenge operator limitations. Sections 3.1.3.3.3 and 3.1.3.4 specify that the M-MIS design process will provide for the assessment of operator workload and identify operator functions and tasks that result from the consideration of the potential for and consequences of failures of plant and M-MIS system components. Section 3.4.1 states that the M-MIS design should not require operators to perform tasks that exceed

their capabilities. Sections 3.7.7.2 and 3.7.7.3 require that maintenance tasks be evaluated to ensure they are within the expected capability of maintenance technicians and that M-MIS equipment be designed to facilitate maintenance and repair and to minimize confusion and chance of error during these operations. The staff concludes that these requirements are consistent with accepted human factors principles and practices and the guidance in Sections 3.4 and B4.2 of NUREG-0700 and are, therefore, acceptable.

3.2.8 Performance of a Comprehensive Task Analysis

Sections 3.1.3.3.2 and 3.7.7.1 of Chapter 10 specify that the design process is to provide for the analysis of operations and maintenance functions and tasks and the validation of the allocation of functions to automatic control systems or particular control stations. The staff concludes that these requirements are consistent with accepted human factors principles and practices and the guidance in Sections 3.4 and B4.3 of NUREG-0700 and are, therefore, acceptable.

3.2.9 Integration of Human Factors Into the Design of Hardware and Software

Section 2.2.8 of Chapter 10 establishes a policy for the application of human factors engineering as a formal part of the M-MIS design and design verification process. The ALWR M-MIS design will place particular emphasis on the elimination of potential sources of human error, the reduction in the probability of human error, and the provision for detection of and recovery from human errors, should they occur. Section 4.1.5 specifies that the design process is to provide for the definition of the specific detailed design practices as they evolve during the design development and that these practices are to be based on published guidance on human factors, such as EPRI NP-3659, "Human Factors Guide for Nuclear Power Plant Control Room Development," and EPRI NP-6209, "Effective Plant Labeling and Coding." In addition, if the designer selects new technology (e.g., "soft controls") for which published guidance on human factors practice is limited, the designer must perform and document limited, special evaluations and reviews to verify that operator performance is not degraded by its use.

Sections 3.1.4.4.1 and 3.1.4.4.2 of Chapter 10 require reviews of the M-MIS functional requirements, system functional designs, and hardware and software specifications to confirm that they are complete, correct, consistent, feasible, and testable. The reviews are to consider system architecture, interfaces, testability, maintainability, reliability, and the human factors aspects of the M-MIS design. Section 3.1.4.4.3 requires that the reviews performed as outlined in Sections 3.1.4.4.1 and 3.1.4.4.2 include an evaluation of the alternatives and tradeoffs considered by the M-MIS designer and others in establishing the requirements and specifications and that they address the correction of system and equipment problems experienced with previous designs. The reviews are to include all aspects of the design including human factors for operators and maintenance personnel. The staff concludes that these requirements are consistent with accepted human factors principles and practices and the guidance in Section 18.1 of the SRP and Section 6 of NUREG-0700 and are, therefore, acceptable.

3.3 Development and Performance of Control Room and Local Control Stations Design Verification

Consistent with the guidance in Section 18.1 of the SRP and Sections 3.7 and B4.4 of NUREG-0700, the staff's position is that the methods and criteria used in the design of the control room and local control stations should conform to accepted human factors principles and practices. For those criteria used for which professional acceptance does not exist, appropriate supporting justification is needed.

Section 4 of Chapter 10 of the passive Requirements Document establishes the requirements for ALWR control stations. Section 4.1 requires the M-MIS designer to establish a process that ensures a consistent design approach for all plant control stations and integrates the identified functions and tasks. Conceptual designs are to be based on the initial definition of tasks and reviewed by an interdisciplinary review team that includes personnel with operational experience, human factors specialists, and persons in the engineering disciplines. Requirements are also established for the use of mockups and active simulators in the review and evaluation of control stations.

Section 4.1.5.1 of Chapter 10 specifies that published guidance on human factors practice such as EPRI NP-3659 be used as guidance on human factors detailed design practices for control station development. In addition, Section 4.1 requires that the M-MIS designer develop the necessary design practices based on the best available information for cases where new technology has been selected and published guidance on human factors is limited. This requirement also specifies that the developed design practice will be verified by experimentation, including live simulation. Experimentation should explicitly be included in the review process with the review team determining the need for further review of design practices by human factors specialists.

Additional requirements in Section 4.1 state that widely used and accepted design practices are to be given preference by the M-MIS designer. However, if these practices are not consistent with accepted human factors guidelines, the M-MIS designer is to perform and document special evaluations and reviews to verify that operator performance is not degraded by their use. Deviations from common design practices and the basis for these differences are also to be documented.

The requirements for certain specific control station systems are given in the following sections of Chapter 10:

- Section 4.3, "Alarms"
- Section 4.4, "Displays"
- Section 4.5, "Controls"
- Section 4.6, "Voice Communication Systems"
- Section 4.7, "Arrangement, Environment, and Equipment
- Section 4.8, "Control Panels
- Section 4.9, "Specific Control Stations" (MCR, local control stations, remote shutdown control stations, emergency response support facilities, TSC, EOF, and data handling and computer facilities)

Generally, requirements for these systems are similar to, and reference, the requirements in Section 4.1 of Chapter 10 such as function and task analysis, testing, evaluation by simulation, and the use of applicable EPRI guidance documents such as EPRI NP-3659. The staff finds these requirements consistent with accepted human factors principles and practices and the guidance in Section 18.1 of the SRP and Sections 3.7, 6, and B4.4 of NUREG-0700. However, the staff's review of these requirements raised the following concerns regarding the intent of some of the more specific requirements in Section 4 of Chapter 10.

Section 4.4.4 of Chapter 10 originally specified that position indication is to be provided for all valves at the location where they are controlled. Yet this same requirement states that position indication for manually operated valves may be provided if required by the function and task analysis. Because the first requirement specified that all valves will have position indication, while the second statement specified that those identified by the function and task analysis may have position indication, these two statements appeared to conflict. Since function and task analysis may not be performed for all human tasks (e.g., maintenance), on the basis of accepted human factors principles and practices and the guidance in Section 6.5 of NUREG-0700, the staff's position is that valve position indication should be provided for all valves, including manual valves, at any location where function and task analysis suggests it is necessary to provide assurance of proper system alignment. By letter dated March 19, 1992, EPRI clarified that there is no intent not to have a position indication at a valve. Chapter 1, Section 12.3.2.3.2 specifically requires that each manual and power-operated valve have a positive, mechanical indication of the valve's overall position. Remote position indication for manually operated valves may be provided if required by the analysis of functions and tasks. The staff concludes that this is acceptable.

In the DSER for Chapter 1 of the evolutionary Requirements Document, the staff identified the issue of acoustical monitoring. Section 8.2.4.4.3 of that chapter states, "The design of the plant shall consider both reduction and attenuation of noise sources to reduce operator noise exposure to levels of Occupational Safety and Health Administration (OSHA) standards." The staff does not agree with this position. OSHA noise levels are the maximum permissible levels of exposure before physical impairment or damage occurs. The noise levels in the control room, remote shutdown panel, and other normally occupied areas should be determined by the accuracy of the communications required within those areas. Areas should be designed so that the acoustic environments will not cause personal injury, interfere with voice or other communication, cause fatigue, or degrade overall human/system interaction effectiveness. For example, within the spaces of the control room, a maximum sound level (A scale) or the corresponding speech interference level could be prescribed. For areas not frequently occupied and where critical conversations and communications do not occur, the A scale sound levels or the corresponding speech interference levels may be significantly higher. The staff concluded that EPRI should identify and describe the acoustical environments in the control room, the remote shutdown panel area, and at local control stations. By letter dated March 19, 1992, EPRI stated that the design of the plant will consider both reduction and attenuation of noise sources to reduce exposure to operations and maintenance personnel. Personnel will be provided with acoustical environments that will not cause personal injury,

interfere with voice or other communications, cause fatigue, or degrade overall system effectiveness. The staff concludes that this is acceptable.

Section 4.5.5 of Chapter 10 specifies that controls of plant components and systems on the MCR workstations normally will be electronic or "soft" controls, such as touch screens or other non-hardwired devices, except where there are specific requirements to the contrary. Where applicable, the soft controls are to meet established human factors guidelines and the requirements of Section 4.5.5. Section 4.1.5.2 requires the M-MIS designer to develop the necessary design practices based on the best available information for new technology, such as soft controls, for which published guidance on human factors practice is limited. Similar statements regarding applicability of design features to meet human factors guidelines are included in Section 4.4.9. In the DSER for Chapter 10, the staff stated that it was not clear what was meant by "where applicable" in Section 4.5.5, and this statement might conflict with the requirements of Section 4.1.5.2. The staff concluded that this was an open issue and that additional information about the use of human factors guidelines for new technology, where applicable, was necessary. This was identified as an open issue. In Revision 3, EPRI reworded Section 4.5.5 to resolve the ambiguity introduced by the term "where applicable." This section now states that where there are established human factors guidelines for "soft" controls, the guidelines will be met. The staff finds the reworded paragraph acceptable. Therefore, this open issue is closed.

Section 4.7.4 of Chapter 10 specifies that control stations will be provided with lighting that can be adjusted by the operators to provide uniform illumination in the range of 10 to 50 foot-candles. In the rationale portion of this section, EPRI references an EPRI pilot research study, EPRI-5989 "Effects of Control-Room Lighting on Operator Performance, a Pilot Empirical Study," which indicated that relatively low lighting levels (10 foot-candles) may be completely adequate and may be preferred by some operators. However, the guidance in Section 6.1.5.3 of NUREG-0700 concerning recommended illumination levels for task performance prescribes minimum levels of 20 to 50 foot-candles and maximum levels of 50 to 100 foot-candles, depending on the type of task. Only where emergency lighting is in use are minimum illumination levels as low as 10 foot-candles recommended. In the DSER for Chapter 10, the staff concluded that the results of a pilot empirical study were insufficient bases for deriving illumination level requirements and identified this as an open issue. The staff will review applications for FDA/DC to ensure that M-MIS designers provide documentation substantiating deviations from NUREG-0700 for levels of illumination in the control room. Therefore, this open issue is closed.

3.4 Development of Plant Procedures

Consistent with the guidance in Section 13.5.2, "Operating and Maintenance Procedures," of the SRP and Section B4.3.2.9 of Appendix B to NUREG-0700, the staff position is that the methods and criteria used to develop plant procedures and operator aids, including software programs, should be in accordance with valid human factors principles and practices. These procedures should provide guidance for plant operation, maintenance, test, and surveillance during all plant operational modes and conditions.

Section 8.2.2.1 of Chapter 1 of the passive Requirements Document requires that procedures and training for operation and maintenance be standardized. Also, a standard set of operating and maintenance procedures and training should be developed for each ALWR design. In addition, standardization of ALWR designs should be addressed to the extent practicable.

Sections 3.1.3.3.3, 3.1.3.5.1, 3.4.2, 3.4.2.1, and 4.1.6.4 of Chapter 10 establish requirements for the development, V&V, and documentation and review of operating procedures. These activities are to be accomplished using the function and task analysis, the plant dynamic model and simulator, and the control station(s) design documentation. The rationale used for these requirements is that procedures are an integral part of the M-MIS design and it is inconsistent with the ALWR goals of improvement of the man-machine interface and standardization to leave the preparation of procedures to individual utilities.

Section 3.1.3.6.2 of Chapter 10 specifies that inservice surveillance testing procedures will be prepared and used in walk-throughs in control station mockups.

Sections 3.4.2.2 and 3.4.2.2.1 of Section 10 require that MCR operator workstations and other workstations utilize electronic, selectable displays and have electronically displayed procedures, where practicable. The M-MIS designer is to establish and document procedure displays and preparation practices and guidelines and use active simulation to validate them. Similarly, Section 3.4.2.3 specifies that hard-copy procedures will be supplied if electronically displayed procedures are not practicable. Preparation practices and guidelines for hard-copy procedures using active simulation are to be established, and procedures are to be consistent in format and content with electronically displayed procedures. The stated rationale for the use of electronically displayed procedures is based on Electricite de France experience at its facilities in France.

The information in Chapter 10 of the passive Requirements Document concerning the use of electronically displayed procedures was not sufficient for the staff to determine if the requirements would be adequate to ensure the development of procedures for safe operation of the ALWR. In its RAI dated April 17, 1991, the staff requested more information on the development of computerized procedures. By letter dated June 13, 1991, EPRI provided the rationale for and advantages of electronically displayed procedures and cited the close relationship between using electronically displayed procedures and providing the procedures as part of the total M-MIS design. In its response EPRI noted that a substantial effort was required of the M-MIS designer to verify and validate electronic procedures and to resolve the concerns of operability and maintenance by active simulation. The V&V process would enable the M-MIS designer to determine the relative merits of the different display formats and integrate the appropriate operating procedures into the M-MIS design. EPRI also discussed the development of hard-copy procedures as necessary as required by Section 3.4.2.3 of Chapter 10 of the passive Requirements Document. EPRI noted that direct comparison of electronically displayed procedures with hard-copy procedures during the design process will determine the relative merits of the two types of procedures, and if electronically displayed procedures are not concluded to represent an improvement, hard-copy procedures will be specified.

The staff has reviewed the EPRI response and concludes that the development of electronically displayed procedures is a desirable goal for the overall integration of operator informational needs into the M-MIS. The staff position is that the M-MIS designer should consider the use of electronically displayed procedures early in the design process to resolve any issues concerning their development, operability, maintainability, and reliability. If electronically displayed procedures are determined to be an improvement over hard-copy procedures and the M-MIS designer has integrated electronically displayed procedures into the overall M-MIS design, they should be provided as part of the design.

Section 3.4.2.2.1 of Chapter 10 requires that the M-MIS designer establish and document procedure preparation guidelines and practices and use active simulation to validate procedure displays. Section 3.4.2.2.2 requires that active simulation be used to determine preferable practices to be followed in preparing electronically displayed procedures. Sections 3.4.2.3, 3.4.2.3.1, and 3.4.2.3.2 require that hard-copy procedures be provided to supplement all electronically displayed procedures, and that hard-copy procedures conform to the same preparation guidelines and be consistent in format and content with electronically displayed procedures. On the basis of the requirements discussed above and the additional information provided in the EPRI response, the staff concludes that the requirement for electronically displayed procedures does not conflict with current regulatory requirements and the guidance for the development of plant procedures in Section 13.5.2 of the SRP and is, therefore, acceptable.

Section 3.4.5 of Chapter 10 requires that the M-MIS design include the definition of the features that will be provided to assist the operator in carrying out specific tasks. Operator aids are to be considered as an integral part of the overall M-MIS design and are to be included in all evaluations of control stations and M-MIS designs.

In the DSER for Chapter 10, the staff stated that Section 3.7.7 of Chapter 10 established the requirements for maintenance human factors but did not include requirements for the development, review, V&V of maintenance procedures as part of the M-MIS design. This appeared inconsistent with the requirements in Chapter 1, Section 8.2.2.1, for the development and standardization of maintenance procedures for each ALWR design. The staff concluded that this was an open issue and that additional information regarding the development, review, V&V of maintenance procedures was necessary. In its response dated February 3, 1992, to the evolutionary plant DSER and subsequently in response to the passive plant DSER, EPRI stated that Section 3.7.7 of Chapter 10 was intended to be consistent with Section 8.2.2.1 of Chapter 1. The staff concludes that the requirements established in the passive Requirements Document for the development of procedures are consistent with accepted human factors principles and practices and the guidance in Section 13.5.2 of the SRP and Section B4.3.2.9 of Appendix B to NUREG-0700 and are acceptable. Therefore, this open issue is closed.

3.5 Personnel Selection and Qualification

On the basis of accepted human factors principles and practices, the staff's position is that the methods and criteria used to select plant personnel, including aspects such as selection criteria, testing, and evaluation techniques, should be based on a systematic function and task analysis.

Section 8.2.6.4 of Chapter 1 contains requirements for personnel and staffing. Generally, the requirements allow the plant designer and constructor to assume that personnel staffing for plant operation and maintenance will be defined very soon after the commitment to build an ALWR. The requirements call for the two top echelons of management to be filled with people with at least 4 years of responsible line-management positions in an operating commercial nuclear power plant. Personnel selected and trained for the plant operating organization are to meet the latest version of ANSI/ANS 3.1, "Selection, Qualification and Training for Nuclear Power Plants." In addition, operations staffing levels specified by EPRI are consistent with the staffing requirements in 10 CFR 50.54(m).

Section 4.2.6 of Chapter 10 of the passive Requirements Document requires that the training, qualifications, and experience of the operations staff members, which will be used in the development of the M-MIS design, be based on current operating practice, where practicable. Additionally, the M-MIS designer is to specify, early in the design process, any levels of training, qualification, and experience of these operations staff members that differ from typical plant owner training and operating practices. These differences and their bases are to be included in the M-MIS design documentation.

Section 3.7.7.1 of Chapter 10 specifies that the M-MIS designer will systematically identify the tasks required to maintain the M-MIS equipment, including, for example, definition of skills, tools, test equipment, and access. The rationale for these requirements is that the plant owner will need to know the qualifications of maintenance personnel and the support and test equipment necessary to maintain M-MIS equipment in operation. Section 3.7.7.4 specifies that M-MIS cabinets will be designed to facilitate access by maintenance personnel. The design of M-MIS cabinets should allow specialized maintenance technicians to work on their particular equipment without interfering with the servicing of other equipment by other technicians.

In the DSER for Chapter 10, the staff concluded that the passive Requirements Document did not specify that plant personnel selection and qualifications, including aspects such as selection criteria, testing, and evaluation techniques, were to be based on a systematic function and task analysis. Rather, selection of both operations and maintenance personnel, including top-level personnel was to be based on the assumed existing nuclear plant owner's practice regarding training, qualifications, experience, and skill levels, which may not be applicable to ALWR designs. The staff identified this concern pertaining to operations and maintenance personnel and top-level personnel as an open issue and stated that additional information regarding the use of system function and task analysis as the basis for the methods and criteria used to select plant personnel was required.

In its response dated January 24, 1992, to the evolutionary plant DSER for Chapter 10 and subsequently in response to the passive plant DSER, EPRI stated

that Section 3.1.3.3.3 of Chapter 10 requires the use of functions and tasks to evaluate operator workloads and the mental and physical burdens of the operators that will be part of the M-MIS design process. EPRI, however, did not require that these results be used as input to specify personnel selection and qualification. EPRI stated that these issues were not considered to be within the scope of the passive Requirements Document. The staff concludes that the resolution of these issues regarding plant personnel selection and qualification is the responsibility of the COL applicant under 10 CFR Part 50 and will be reviewed by the staff at the COL stage. The staff finds this position acceptable. Therefore, this open issue is closed.

3.6 Personnel Training and Testing

Consistent with the guidance in Section 13.2, "Training," of the SRP and Section B4 of Appendix B to NUREG-0700 and accepted human factors principles and practices, the staff position is that the methods and criteria used to train plant personnel and evaluate their job performance, including simulator training, should be based on a systematic function and task analysis. Elements of the human factors program for personnel training and testing should include, at a minimum, the following:

- job-specific analysis
- description of standardized training programs
- identification of approaches and applications of training to be conducted in conjunction with onshift activities or ancillary training
- development of a standardized plant reference training facility
- suggested site functional organization including interfaces
- utility/licensee interfaces for the organizational structure and composition of plant staff

Section 8.2.2.1 of Chapter 1 of the passive Requirements Document requires that training for operations and maintenance be developed and standardized for each ALWR. Section 8.2.6.4 of Chapter 1 requires that appropriate personnel from the plant operating organization participate in an owner-developed training program. The training program is to provide knowledge of the design basis of the plant and the implementation plan for the project. In addition, personnel selected and trained for the plant operating organization are to meet the requirements of the latest version of ANSI/ANS 3.1, "Selection, Qualification and Training for Nuclear Power Plants."

Section 3.1.3.3.3 of Chapter 10 specifies that the M-MIS design process will provide for the use of identified functions and tasks in the development of training requirements and V&V reviews.

Section 3.4 of Chapter 10 requires the M-MIS designer to explicitly consider the actions of the operators and other members of the plant staff when operating and controlling the plant. These actions are to be within the capability of all operators. Section 3.4.1.1 requires that the M-MIS design not require the operators to perform tasks that overburden their capabilities

and that compliance with this requirement should be confirmed by dynamic simulation of the tasks in a full-scope simulator. In addition, the M-MIS designer is to specify which operator workload measures are to be used and provide justification for their use.

Sections 3.7.7.1 and 3.7.7.2 of Chapter 10 specify that the M-MIS designer will systematically identify the tasks required to maintain the M-MIS equipment including, for example, definition of skills, tools, test equipment, and access. The M-MIS designer should evaluate the maintenance tasks to ensure that the required maintenance actions are simple, well understood, and within the expected capability of maintenance technicians. The evaluations should include the use of mockups or prototypes of typical M-MIS equipment and the performance of maintenance task walk-throughs.

Section 4.2.1.1 of Chapter 10 requires that the M-MIS designer specify the responsibilities assumed in the design for each member of the operating crew. This includes responsibility for supervision and consideration of all plant operating modes and conditions. This information will be needed by the plant owner for staffing the plant and planning for training.

Section 4.2 of Chapter 10 contains requirements for the analysis and evaluation of operator and maintenance tasks and operator workload. The results of these analyses and evaluations are to be reviewed and verified on mockups, prototypes, and simulators. A requirement for the specification of the operating crew and supervisory responsibilities to be used for plant staffing and training planning is also included. In the DSER for Chapter 10, the staff concluded that these requirements were consistent with accepted human factors principles and practices and were, therefore, acceptable.

However, the passive Requirements Document did not include requirements for (1) descriptions of standardized training programs, (2) the identification of approaches and applications of training to be conducted in conjunction with onshift activities or ancillary training, (3) the development of a standardized plant reference training facility, (4) a suggested site functional organization including interfaces, and (5) utility/licensee interfaces for the organizational structure and composition of plant staff. In its response dated January 24, 1992, to the evolutionary plant DSER for Chapter 10 and subsequently in response to the passive plant DSER, EPRI stated that these issues were not considered to be within the scope of the passive Requirements Document. The staff concludes that the resolution of these issues regarding the top-level training requirements of Sections 8.2.2 and 8.2.6.4 of Chapter 1 of the passive Requirements Document is the responsibility of the COL applicant under 10 CFR Part 50 and will be reviewed by the staff at the COL stage.

3.7 Human Factors Tests and Evaluations

3.7 Section B4.4 of Appendix B to NUREG-0700 and accepted human factors principles and practices form the basis for the staff's position that human factors tests and evaluations should be performed as part of an integrated effort within the total test and evaluation program. Human factors test and evaluation activities should include, at a minimum, the following elements:

- development of a human factors V&V test plan

- a method for documenting test activities to provide traceability and to ensure that all human factors requirements are addressed during test and evaluation

- testing of a fully operational control room prototype to determine if the performance objectives of the plant can be met given the equipment design, software design, procedures, training, and organization/staffing complement

- development of quantitative measures to assess human-system performance

- description of how the program will determine if undesirable design or procedural features have been introduced during the design process

- a plan for resolving problems uncovered during the test phase

The passive Requirements Document contains requirements addressing several of these elements. The staff's evaluation of EPRI's requirements for those elements is presented below.

3.7.1 Development of a Human Factors Verification and Validation Test Plan

The original version and Revision 1 of Chapter 10 of the passive Requirements Document did not include a requirement for the development of a human factors V&V test plan. In response to a request for additional information, EPRI, in a letter dated December 6, 1990, committed to add to Section 11.8.5 of Chapter 1 a requirement that the ALWR design development plan specifically identify the features that will ensure a systematic consideration of human factors in the design process. The requirement will specify that the plant designer is to use EPRI NP-3659 and NP-4350 as guidance to determine the features that must be included in the design development plan to address human factors. However, as stated in the DSER for Chapter 10, it was not clear if this requirement would include the development of a human factors V&V test plan. The staff concluded that EPRI should clarify if Section 11.8.5 of Chapter 1 of the passive Requirements Document would include requirements for the development of a human factors V&V test plan and identified this as an open issue. In its response dated February 3, 1992, to the evolutionary plant DSER and subsequently in response to the passive plant DSER, EPRI stated that it was its intent that the M-MIS test plans required by Section 3.1.3.6.1 of Chapter 10 include testing to ensure that V&V of human factors considerations have been properly considered in the design. Changes have been made to Section 11.8.5 of Chapter 1 and Section 3.1.3.6.1 of Chapter 10 to reflect EPRI's intent. The staff finds that these changes provide adequate clarification and are, acceptable. Therefore, this open issue is closed.

3.7.2 Documentation of Test Activities

Section 3.1.3.5.2 of Chapter 10 requires that the M-MIS design process include the development of digital computer-based dynamic models for the overall plant responses as well as individual control systems, including operator actions. The plant designer will document these dynamic models and provide the documentation as part of the final M-MIS design.

Sections 3.1.4.4.1 and 3.1.4.4.2 of Chapter 10 require that the review team include in its review the M-MIS system functional requirements, system functional designs, and the hardware and software specifications. The review team is also required to confirm and concur that the functional design and specifications will meet plant system functional requirements and will result in a satisfactory implementation of the functional design. These reviews will also consider human factors aspects of the design. In the evolutionary plant DSER for Chapter 10, the staff concluded that EPRI should clarify if these requirements were intended to (1) include a method of documenting human factors test activities to provide traceability and (2) ensure that all human factor requirements are addressed during test and evaluation, and identified this as an open issue. In its response dated February 3, 1992, to the evolutionary plant DSER and subsequently in response to the passive plant DSER, EPRI stated that the review team will document human factors tests in the same manner as any other testing to support the M-MIS design and will document and reference the results so that the basis of the design can be traced. In addition, EPRI requires that all human factors requirements are addressed during test and evaluation. EPRI modified Sections 3.1.4.4.4 and 3.1.4.4.5 of Chapter 10 to reflect this position. The staff concludes that these changes address its concern about documentation of the human factors tests and scope of test and evaluation and are acceptable. Therefore, this open issue is closed.

3.7.3 Testing of a Fully Operational Control Room Prototype

Section 11.10.4 of Chapter 1 specifies that the plant designer is to provide a plant simulator/performance model that can be used as a design tool in studying plant responses and human-engineering aspects of the plant controls and control room design and in developing plant operating procedures for normal, abnormal, and accident events.

Section 3.1.3.5.2 of Chapter 10 requires that the M-MIS design process include the development of digital computer-based dynamic models for the overall plant response as well as individual control systems, including operator actions. These dynamic models will be

- suitable for analyzing both steady-state and transient behavior
- used to confirm the adequacy of control schemes
- used to confirm the allocation of control to an automatic system or an operator
- used to develop and validate plant operating procedures
- validated against tests of actual plant behavior wherever practicable
- developed early enough in the design process that modifications of the systems themselves can be made, if shown to be needed by the analysis
- incorporated, as directly as possible, into plant general-purpose or limited-use simulators

- completely documented and the documentation provided as part of the final M-MIS design

The staff concludes that these requirements are consistent with the guidance in Section B4 of Appendix B to NUREG-0700 and accepted human factors principles and practices and are acceptable.

3.7.4 Development of Quantitative Measures To Assess Human-System Performance

Sections 3.4.1.1 and 3.7.7.2 of Chapter 10 require that the M-MIS design not require operators and maintenance personnel to perform tasks that would overburden their capabilities. Compliance with these requirements will be confirmed by dynamic simulation of the operators' tasks in a full-scope simulator and performance of maintenance task walk-throughs on mockups or prototypes of typical M-MIS equipment.

Section 3.4.1.3 of Chapter 10 requires that the M-MIS design include features that support and facilitate a team approach. These include features that will enhance team communications and cohesion and provide the capability for team members to support and back each other up. These features should be based on information available when the M-MIS is designed. The basis for these features should be part of the M-MIS design documentation.

Section 4.9.3.8 of Chapter 10 states that any action taken to enable a remote shutdown station or transfer of control to it will be annunciated in the MCR. The M-MIS designer should determine, using analysis and active simulation, if the annunciation in the MCR will provide adequate time for operators and plant security personnel to take action to prevent an accident in the case of unauthorized use of a remote shutdown station.

Section 4.9.4.1.1 of Chapter 10 establishes the requirements for access between the MCR and the TSC. The intent of these requirements is to facilitate easy access between the MCR and the TSC during emergencies, put less reliance on voice communications, and discourage people from congregating in the MCR and disrupting operations. Easy access between the two spaces will facilitate the support of the operators and the team approach to coping with emergencies.

In the DSER for Chapter 10, the staff stated that the EPRI requirements appeared to focus on design considerations and limited assessments and tests that address operator performance and maintainer tasks. Team performance was only addressed as a feature that must be considered during the design process. There appeared to be no requirement for the development of quantitative measures to assess team-system performance, that would include, for example, verification that the expected team-system performance interaction between MCR operators and personnel staffing the TSC and EOF was achieved. The staff concluded that additional information concerning requirements to assess team performance was necessary and identified this as an open issue.

In response to the November 3, 1992, meeting between the staff and EPRI, Sections 2.2.8, 3.4.1.1 and 4.3.1.4 of Chapter 10 were shown to include the requirements that testing and validation of the M-MIS include both qualitative and quantitative measures of human performance; that the designer specify in the design plan the operator workload measures to be used; and that specific

measures be defined for measuring alarm system effectiveness, respectively. In addition, Section 4.1.3.3 was revised to ensure that evaluations using simulators will be based on appropriate qualitative and quantitative measures, and specifies that evaluations involving operating teams or crews use appropriate qualitative and quantitative measures of team performance. The staff concludes that the changes to the passive Requirements Document satisfy the issues raised with respect to the quantification of human performance and, are therefore, acceptable.

3.7.5 Determination of Undesirable Design or Procedural Features and Resolution of Problems Uncovered During the Test Phase

Sections 3.1.3.5.2, 3.1.4.4.1, 3.1.4.4.2, and 3.1.4.4.6 of Chapter 10 require the development of digital computer-based dynamic models for overall plant response as well as individual control systems, including operator actions, the review and documentation of the functional requirements, and the human factors aspects of the hardware and software specifications. Sections 4.1.3.2 and 4.1.3.3 require that during the design review process, the fabrication of each control station and the use of active simulation of the control station be evaluated. Fabrication of the control station will evaluated early in the design review process so that the results of the evaluations can be used to modify the M-MIS and plant systems designs. Section 4.1.4 requires that the control station design review process provide for the iteration of functions and tasks assigned to the control station for feedback regarding all aspects ranging from the design of the individual control stations to the overall identification of functions and tasks and their assignment to particular control stations. The rationale for this requirement is to provide a means for difficulties found in the course of the review to be resolved by reassignment of tasks. The staff concludes that these requirements are consistent with the guidance in Section 18.1 of the SRP and Sections 3.6, 3.7, 3.8, 3.9, 4, and B4.4 of NUREG-0700 and accepted human factors principles and practices and are acceptable. Therefore, this open issue is closed.

3.7.6 Human Factors Open Issues Originating From Other Chapters

In the DSER for Chapter 1, the staff recommended that EPRI include IEEE P1023/D5, "Guide for the Application of Human Factors Engineering to Systems, Equipment and Facilities of Nuclear Power Generating Stations," as a reference. It also recommended EPRI NP-2360, "Human Factors Methods for Assessing and Enhancing Power Plant Maintainability," as an excellent reference. The staff identified referencing of these documents as an open issue. In its response dated May 17, 1991, to the DSER for Chapter 1, EPRI stated that Chapter 6, Section 2.1.1, referred to EPRI NP-4350, "Human Engineering Design Guidelines for Maintainability." However, it did not state that it intended to include the IEEE standard as a reference. It is the staff's position that IEEE P1023/D5 should be included as a reference document because, unlike EPRI reports, IEEE documents are periodically updated. Therefore, the applicant for FDA/DC should include IEEE P1023/D5 as part of its design reference documentation. This DSER open issue is closed.

4 CONCLUSIONS

The staff concludes that the human factors requirements in the EPRI passive Requirements Document do not conflict with current regulatory requirements and

guidelines. They are, therefore, acceptable. However, by themselves they do not provide sufficient information for the staff to conclude that an ALWR applicant referencing the passive Requirements Document will adequately incorporate all human factors issues related to ALWR plant design in a manner that will achieve increased safety and reliability and ultimately reduce the probability of human error. Therefore, applicants referencing the passive Requirements Document will be required to provide sufficient information to demonstrate that their human factors program will ensure that the level of human performance required to maintain plant safety is achieved.

The staff concludes that the EPRI Requirements Document for passive plant designs specifies human factors requirements that, if properly translated into a design in accordance with the NRC regulations in force at the time the design is submitted, can result in an acceptable nuclear power plant design.

Chapter 11, "ELECTRIC POWER SYSTEMS"

1 INTRODUCTION

This chapter of the SER documents the NRC staff's review of Chapter 11, "Electric Power Systems," of the passive Requirements Document through Revision 5. Revision 5 was submitted to the NRC by letter dated October 1, 1993. Chapter 11 was prepared, under the project direction of EPRI and the ALWR Utility Steering Committee; by Commonwealth Edison Company; Duke Power Company; General Electric Company; MPR Associates, Inc.; S. Levy Incorporated; Sargent and Lundy Engineers; Science Applications International Corporation; Westinghouse Electric Corporation; and EPRI.

1.1 Review Criteria

Section 1 of Volume 1 of this report describes the review criteria used by the staff during its review of Chapter 11 of the passive Requirements Document.

1.2 Scope and Structure of Chapter 11

Chapter 11 of the passive Requirements Document defines the ALWR Utility Steering Committee's requirements for the design of the systems that will supply electric power to the plant auxiliary and service loads, including instrumentation and control system loads. They consist of the offsite power system, the medium- and low-voltage onsite ac power distribution systems, the onsite standby ac power supply system, the dc and low-voltage vital ac power supply systems, and the lighting and electrical protective systems.

1.3 Policy Issues

During its review of Chapter 11 of the passive Requirements Document, the staff did not identify any issues that involve policy questions for the technical areas discussed in this chapter, other than those already identified in Commission papers listed in Appendix B to Chapter 1 of this report.

1.4 Outstanding Issues

The DSER for Chapter 11 of the passive Requirements Document contained the following outstanding issues:

Open Issues

P.11.0-1 regulatory treatment of active non-safety systems (2.2.1, 2.2.2, 3.2.2, 4.2.1, 4.2.2, 5.2.1, 5.2.4, 5.2.5, and 8.2.1)

P.11.0-2 clarification of terminology describing the role of non-safety systems (2.2.1)

P.11.0-3 loss of non-safety ac power during shutdown conditions (2.2.6)

P.11.0-4 second offsite power supply circuit for permanent non-safety load buses during power operations (3.2.1)

P.11.0-5 consistency of Appendix A to Chapter 1 assumptions with Chapter 11 (3.2.3)

P.11.0-6 data provided in Appendix A to Chapter 1 relative to loss of offsite power events (3.2.5)

P.11.0-7 non-safety ac electrical power systems during a small-break loss-of-coolant accident (5.2.1)

P.11.0-8 clarification of revised requirements in Sections 1.5.2 and 5.3.1.1 (5.2.1)

P.11.0-9 applicability of the 2000-hour rating and its impact on peaking operation (5.2.1)

P.11.0-10 peaking operation of non-safety standby power sources (5.2.3)

P.11.0-11 consistency of failure and unavailability rates in Appendix A to Chapter 1 (5.2.4)

P.11.0-12 justification for the use of independent self-contained cooling systems (5.2.6.3)

P.11.0-13 potential loss of dc buses (7.2.1)

P.11.0-14 load shedding for 72-hour battery endurance (7.2.3)

P.11.0-15 backup battery and battery charger (7.2.4)

P.11.0-16 transfer scheme for the safety dc power supply system (7.2.5).

P.11.0-17 battery and battery charger instrumentation and alarms (7.2.8)

P.11.0-18 lack of requirements for electric protective assemblies in reactor protection system power for BWRs (7.2.9)

Confirmatory Issues

P.11.C-1 isolation of safety systems from their non-safety sources (2.2.5)

P.11.C-2 minimum starting voltages for valve actuator motors (2.2.8)

P.11.C-3 requirements for main step-up transformers (3.2.6)

P.11.C-4 harmonic distortion effect of adjustable speed drives (4.2.3)

P.11.C-5 location of non-safety standby ac power sources (5.2.2)

P.11.C-6 sizing of thermal overload devices (6.2.1)

P.11.C-7 battery sizing criteria specified for 72-hour coping capability (7.2.2)

P.11.C-8 physical and electrical separation of the safety dc and low-voltage vital ac power supply system (7.2.5)

P.11.C-9 non-safety dc power supplies for switchyard circuit protection and control equipment (7.2.6)

P.11.C-10 battery capacity margin (7.2.7)

P.11.C-11 normal and emergency lighting following design-basis events (8.2.1)

P.11.C-12 intensity of the emergency lighting system (8.2.2)

The final disposition of each of these issues is discussed in detail in the appropriate section of this chapter, as indicated by this parenthetical notation following each issue. All outstanding issues identified in the DSER for Chapter 11 have been resolved.

1.5 Vendor- or Utility-Specific Items

The DSER for Chapter 11 contained the following vendor- or utility-specific items. The final disposition of each item is discussed in the appropriate section of this chapter, as indicated by the parenthetical notation following each item.

P.11.V-1 reliance on the non-safety electrical systems beyond 72-hour period (2.2.2)

P.11.V-2 reliance on the non-safety electrical systems to achieve cold shutdown (2.2.2)

P.11.V-3 applicable regulations and regulatory guidance that are not addressed in Chapter 11 (2.2.3)

P.11.V-4 minimization of Class 1E components (2.2.4)

P.11.V-5 use of revisions to Institute of Electrical and Electronics Engineers standards not endorsed by the staff (2.2.7)

P.11.V-6 operating conditions of all plant loads for all relevant grid conditions and the design of the bus voltage protection schemes (3.2.4)

P.11.V-7 design of the standby power source starting system (5.2.6.1)

P.11.V-8 design of the standby power source fuel oil storage and transfer system (5.2.6.5)

P.11.V-9 design of the electrical separation of dc and vital ac power supply systems (7.2.5)

P.11.V-10 integration of the exterior lighting system with the closed-circuit television system (8.2.4)

The following vendor- or utility-specific items remain and will be addressed during the staff's review of an individual application for FDA/DC. The staff's position on each item is provided in the appropriate section of this chapter, as indicated by the parenthetical notation following each item.

P.11.V-1 regulatory treatment of non-safety systems (2.2.1, 2.2.2, 2.2.3, 2.2.6, 3.2.1, 3.2.2, 5.2.1, 5.2.3, 5.2.4, 5.2.6, and 8.2.1)

P.11.V-2 electrical systems requirements for severe-accident management (2.2.2)

P.11.V-3 minimization of Class 1E components (2.2.4)

P.11.V-4 protective devices for chargers and transformers for Class 1E batteries (2.2.5)

P.11.V-5 use of latest version of the IEEE Standard (2.2.6)

P.11.V-6 consistency of probabilistic risk assessment (PRA) assumption with Chapter 11 (3.2.3)

P.11.V-7 analyses of plant loads under relevant grid conditions (3.2.4)

P.11.V-8 use of loss of offsite power data in PRA models (3.2.5)

P.11.V-9 technical specification limitations for dc and low-voltage ac power supply systems (7.2.1)

P.11.V-10 load shedding for 72-hour battery endurance (7.2.3)

P.11.V-11 effects of low battery voltage (7.2.3)

P.11.V-12 battery discharge profiles (7.2.3)

P.11.V-13 details of design for electrical separation (7.2.5)

P.11.V-14 battery and battery charger instrumentation (7.2.8)

P.11.V-15 alarm systems for BWR scram solenoids (7.2.9)

P.11.V-16 integration of exterior lighting systems with the closed-circuit television system (8.2.4)

2 POLICY STATEMENTS AND GENERAL REQUIREMENTS

2.1 Policy Statements

Section 1.5 of Chapter 11 of the passive Requirements Document identifies policy statements established by the ALWR Utility Steering Committee that form the basis for the specific design requirements in Chapter 11. Section 1.5.1 of Chapter 11 states that the overall objective of the design features specified in this chapter is to achieve the goals described in Section 1 of Chapter 1 of the passive Requirements Document.

Section 1.5.2 of Chapter 11 states that, since electric power is necessary to operate most auxiliary and service systems, the availability and reliability of the electric power systems are essential to achieving the safety and availability goals established for the ALWR passive plants. The section goes on to discuss the availability and reliability aspects of the onsite and offsite electric power systems. In that context, the section discusses the lack of Class 1E qualification for the standby ac power sources (which, for example, may be diesel generators or combustion turbine-generators), the redundancy of the standby ac power sources, and the lack of a requirement for the reserve offsite power supply circuit. This evaluation addresses these issues in the succeeding sections that deal specifically with the standby ac power sources and the offsite power system.

Section 1.5.3 of Chapter 11 discusses loss of ac power and station blackout. It states that the passive safety systems are designed to perform their safety functions for at least 72 hours following the initiation of a station blackout event. Although independent from ac motive power, the passive safety systems depend on power supplied by batteries to initiate and control automatic operation.

Section 1.5.4 of Chapter 11 states that simplification of systems relative to current plant designs is a fundamental principle of the ALWR program, intended to enhance plant safety, reliability and availability, and constructability. It further declares that, as it applies to the electric power systems, this principle must lead to simple system configurations that will use the minimum number of components and interconnections required to provide the backup and redundancy features needed for safety and availability purposes.

Section 1.5.5 of Chapter 11 states that development of standardized nuclear power plant designs is viewed as an effective means of achieving cost reduction and regulatory stabilization; to support the standardization efforts, Chapter 11 stipulates specific configuration requirements for the electric power systems, as has been done for the other systems of the ALWR passive plant. The intent is to limit changes to the general configurations specified in Chapter 11 to only those forced by specific, unusual site conditions. The "reference" plant used to develop the requirements for the electric power systems, as well as the other plant systems, is a single-unit plant with an output capacity of 600 MWe. EPRI states that, for a plant built at an existing site, it is expected that some interconnections not considered in Chapter 11 will be provided between the offsite power system of the ALWR unit and that of other generating units at the same site to further improve the overall reliability of the power supply.

2.2 General Requirements

The requirements specified in Section 2 of Chapter 11 of the passive Requirements Document are requirements that apply to the integrated set of electric power systems of the ALWR passive plant.

Section 2.3.2 of Chapter 11 calls for the use of a three-tier concept in the arrangement of the onsite power distribution systems. The first tier will consist of two divisions of non-safety ac power distribution systems that will feed non-safety loads required exclusively for unit operation. The second tier will consist of the non-safety ac and dc power distribution systems that will supply power to "permanent non-safety loads," that is, non-safety loads that because of their specific functions, are generally required to remain operational at all times or when the unit is shut down. The second tier will also be divided into two divisions and will be provided with redundant onsite standby power sources. The second-tier ac power distribution systems and the "permanent non-safety loads" that they will feed will, to a large extent, consist of electrical systems and loads that have traditionally been classified as safety grade in past non-passive plant designs.

The third-tier distribution systems will consist of the safety dc power distribution system and the associated low-voltage vital ac power distribution systems that will feed the safety (Class 1E) loads. The third-tier distribution system will be divided into independent divisions consistent with the number of redundant load groups they supply. Each division of the third-tier safety power distribution systems will be provided with physically separated and electrically independent batteries sized to supply emergency power to support the passive engineered safety systems in the event of a loss of all other power sources. In addition, each division will include provisions for plug-in connections of transportable ac power generators capable of supplying power to the safety loads.

Section 2.3.10 of Chapter 11 states that the sets of circuits that will constitute the divisions of the safety power distribution systems will be physically separated and electrically independent. Independence and separation will be maintained throughout the load groups, and no cross-ties will be used between buses and circuits belonging to different safety divisions. Section 2.3.11 of Chapter 11 states that non-safety circuits are required to be physically separated from safety circuits throughout the plant. Section 2.3.9 states that non-safety circuits are not permitted to be connected to safety circuits or power sources with the exception of the non-safety power supply circuits that will be connected to the input terminals of the Class 1E distribution systems.

Section 2.6.1 of Chapter 11 specifies that electric power systems are required to be designed for a 60-year operating life without replacement of major components or cabling. However, the design of the systems and the building arrangement are required to permit such replacement, if needed. In the DSER for Chapter 11, the staff determined that the provisions for a 60-year operating life were acceptable, pending the resolution of the open issues regarding equipment qualification identified in Chapter 1 of the DSER.

2.2.1 Regulatory Treatment of Non-Safety Systems Relative to the Use of Non-Safety AC Power Systems

In contrast to both the current generation of light water reactors and the evolutionary ALWRs, the passive ALWR designs rely on safety systems that will use the driving forces of buoyancy, gravity, and stored energy sources. These passive systems will supply safety injection water, provide core and containment cooling, and perform other functions. There will be no pumps in the passive safety systems, and all valves will be actuated by dc electric power from batteries, will be air operated, or will use check valves actuated by the pressure differential across the valve. EPRI proposes that these designs do not include safety-grade ac electric power, other than low-voltage vital ac power supply systems (inverter-supplied).

The passive Requirements Document, however, also includes non-safety-grade active systems to provide defense-in-depth capabilities for reactor coolant makeup and decay heat removal. These systems will serve as the first line of defense in the event of transients or plant upsets to reduce challenges to the passive systems. In addition, the passive ALWRs will also include non-safety-grade active systems, such as the control room heating, ventilating, and air-conditioning systems, for mitigating the radiological consequences of an accident. Many of these systems traditionally have been safety-grade systems and were powered by safety-grade ac power systems with emergency power supplied from safety-grade ac standby power sources (diesel generators). In the passive plants, neither the standby ac power sources nor the ac power systems will be designed to meet safety-grade criteria.

Although the passive safety systems will be designed to meet design-basis acceptance criteria without reliance on the non-safety support systems, the staff stated in the DSER for Chapter 11 that it was concerned that uncertainties remained relative to the performance of the unique passive features and overall performance of core and containment heat removal because of the lack of proven operational history. For example, there were uncertainties about the performance of check valves in the passive safety systems that will operate at low differential pressures provided by natural circulation and gravity injection, as compared to the higher differential pressures provided by a pumped system. These uncertainties increase the importance of the active non-safety systems in providing the defense-in-depth to prevent and mitigate accidents and core damage. As a result, the staff identified an open issue for the passive ALWR plants pertaining to the regulatory treatment of non-safety systems (RTNSS). Because the use of the non-safety-grade ac distribution systems and non-safety ac standby power sources is a part of this issue, the staff indicated in its DSER that the final resolution could affect staff decisions made in this review of Chapter 11. The staff indicated that it would, therefore, report on the final resolution of the non-safety system issues in its final SER.

In its letter dated May 26, 1993, EPRI proposed an approach to determine which systems should receive what kind of regulatory oversight. In Revision 6, EPRI's requirements for implementing this process were provided in Section 1.6 of Appendix A to Chapter 1 of the passive Requirements Document. A key element of this process is to ensure that the passive plant vendors use probabilistic risk assessment methods to examine the reliability of the safety and non-safety systems and ensure that important risk insights are considered

in the design. This process, and the staff's evaluation of it, are described in greater detail in SECY-94-084, which is provided in Annex B of Appendix B to Chapter 1 of this report. Also in SECY-94-084, the staff recommended that the RTNSS process be used to resolve the ac power system issue on passive ALWRs. In its SRM dated June 30, 1994, also provided in Annex B, the Commission approved the staff's recommendations on RTNSS as well as the recommendation that the RTNSS process be used to resolve the ac power systems issue. The staff will review implementation of EPRI's process during its review of an individual application for FDA/DC to ensure that the non-safety-grade ac distribution systems and non-safety-grade ac standby power sources are included in the RTNSS process.

In the DSER for Chapter 11, the staff noted a related concern. It stated that the passive Requirements Document used terminology that had very specific regulatory meanings when used in the context of traditional light water reactors. For example, terms such as "single failure" and "active and passive failures" have been treated in the general design criteria (GDC) and industry standards such as those of the Institute of Electrical and Electronics Engineers (IEEE). In the context of those requirements, these terms have specific meanings for use in evaluating safety-related systems. It was clear that EPRI was not always using the traditional interpretation of this terminology. For example, regarding electric systems, traditionally, there has been no differentiation between active and passive failures (as compared with fluid systems which have been differentiated) in applying the single-failure criterion (SFC). There was no need to differentiate because "single failure of passive components in electric systems should be assumed in designing against a single failure" was stated in the definitions included with the GDC. Therefore, electrical failures were not classified as active or passive. Another potentially more important example involved the application of the SFC. The traditional application of the SFC to electric systems involved only safety-related electric equipment and carried with it many underlying principles such as detectability of failure (via periodic testing), equipment quality, functional as well as physical independence of redundant equipment, and other features.

Therefore, caution needed to be used in reading and understanding the statements made in the electrical-related sections of the passive Requirements Document and this SER. Clarity in the use of such terminology may become more critical as the role of the non-safety systems is further established, because it may become desirable to apply some of the traditional safety system principles and design criteria to the passive plant "non-safety systems." The staff's position was that EPRI should clearly describe the criteria applicable to the non-safety systems using terminology that would not cause confusion because of traditional uses of the terminology for safety systems. The staff identified this as an open issue.

In its letter dated September 29, 1992, EPRI responded to this issue, providing a clarifying change to the rationale of Section 2.3.4 in Chapter 11 that reads in part as follows:

> As required in Paragraph 2.3.3.2 of Chapter 3, redundancy is intended to ensure that each non-safety auxiliary system important to defense in depth and investment protection can be maintained functional in the event of the most limiting single active failure.

> Correspondingly, the non-Class 1E electrical systems that supply power to those systems shall be designed to maintain their supporting function in case of the most limiting single active failure, ... The non-Class 1E systems shall also maintain their supporting capability in the case of the most limiting loss of design function capability of any single passive electric component such as a bus, a cable, a connector, a terminal, etc. . . .

The above indicates that EPRI's intent is to support the redundant non-safety auxiliary systems identified in Section 2.3.3.2 of Chapter 3 with redundant electrical systems that will maintain their support capability given a loss of function of any one of their components. EPRI also committed to a clarifying change to Section 2.3.1.4 in Chapter 3. These changes have all been incorporated by Revision 4 into the passive Requirements Document. The staff finds that these changes satisfactorily clarify EPRI's requirements on this issue and resolve the DSER open issue.

2.2.2 Utilization of the Non-Safety AC Power Systems Before and After 72 Hours of Passive Safety System Operation

One of the major differences between the passive ALWR and existing plants in the electrical area is that the major portion of the ac electrical distribution system and its power sources and support systems are designed as non-safety. (Note: The exception is the vital ac systems, which will be safety related.) The rationale for the difference is that the passive safety systems will be designed to provide the required reactor coolant system (RCS) inventory makeup and heat-removal capabilities, as well as maintain containment integrity, for 72 hours with no reliance on ac power given a loss-of-coolant accident (LOCA) and/or loss of the non-safety heat removal systems. There was, however, little discussion in the passive Requirements Document on the need for non-safety ac power following the 72-hour period of passive safety systems operation; what there was, was sometimes contradictory. The staff assumed that, as a minimum, ac motive power would be needed beyond 72 hours to power the non-safety decay heat removal systems or provide makeup to passive safety system pools; or to power heating, ventilating, and air conditioning systems to maintain long-term instrumentation, equipment, and habitability environments; or to recharge batteries and power the dc electrical systems. The non-safety ac electrical power systems; however, may not have the same degree of protection against fires, flooding, missiles, seismic events, tornadoes, pipe ruptures, and single failures as do safety systems (e.g., Section 2.3.13 of Chapter 11 specifies that non-safety equipment and circuits pertaining to one of the two non-safety divisions will be physically separated to the "maximum extent practical" from equipment and circuits pertaining to the other non-safety divisions). As a result, there is a greater potential that events requiring the use of the passive safety systems will be accompanied or initiated by a loss of the non-safety ac electrical power systems; in which case, the non-safety ac power systems may not be available beyond 72 hours.

Section 1.2.1.1 of Chapter 5 also states that, in order to keep passive systems in a state of preparedness at all times, it is acceptable to utilize support systems with active components such as ac-powered pumps and valves.

It is, therefore, apparent that the non-safety ac systems will also be used to support passive safety systems before events requiring operation of the passive systems.

It is essential that there be a clear understanding of the role of the non-safety electrical systems 72 hours both following an event and preceding an event, and how their availability (or the availability of an alternate system) will be ensured at these times. It is also essential that there be a clear understanding of the difference between the onsite non-safety electrical power systems specified for the passive ALWR and the onsite safety-related electrical power systems found in existing plants. The staff, therefore, requested that EPRI address this issue separately, identifying the minimum non-safety electrical ac power system loads (or their alternates) that are required to be available 72 hours following an event or preceding an event as support systems for the passive safety features. It also asked that EPRI compare each design feature of the non-safety ac electrical power system to each feature of a safety-grade ac electrical power system, identifying the similarities and differences, and demonstrating that the appropriate level of capability is specified in the passive Requirements Document in order to ensure the availability of ac power to the minimum required post-event loads and minimum required pre-event loads. If transportable ac power sources (other than those to the safety loads) are intended to be used, EPRI was asked to discuss the provisions made for their availability and their connection to the non-safety distribution systems as well as provisions taken to ensure that the onsite non-safety distribution systems themselves are not damaged.

In a letter dated January 10, 1992, EPRI responded to this issue. It stated that passive plant designs will not require non-safety ac power to meet licensing design-basis (LDB) events, such as a LOCA and/or loss of the non-safety heat removal systems. EPRI stated that during the specified 72-hour period, passive safety systems will provide decay heat removal and maintain the plant in a safe shutdown condition, including maintaining core and containment monitoring capabilities; only simple, unambiguous actions and easily accomplished offsite assistance will be necessary beyond 72 hours to prevent fuel damage. EPRI further stated that the passive plant designs will utilize safety-grade onsite facilities and connections to be used in conjunction with offsite equipment, as necessary, to sustain the safety functions in the post-72-hour period, and offsite equipment used for these purposes will be readily available and transportable to the site.

In its response, EPRI sent revised Sections 2.2.10 and 2.2.11 of Chapter 5 of the passive Requirements Document that it indicated would help clarify the requirements above. Revised Section 2.2.10 requires that for LDB events, the passive engineered safety systems have the capability to perform their safety functions and maintain the plant in a safe state with the initial inventory of coolant, air supply, battery capacity, and heat sinks of the safety-grade systems, and without credit for non-safety systems, for at least 72 hours. It further requires that the passive safety systems continue to be available for an extended period to perform their safety function by use of onsite facilities and by connecting readily available offsite equipment and resources to replenish these inventories and provide support as may be necessary. The onsite facilities and the connections to utilize offsite support are required to be designed to be readily accessible and to withstand safe shutdown earthquake (SSE) conditions. Offsite equipment and resources required to

maintain the plant in a safe state (e.g., for continued operation of the passive safety systems) must be shown to be readily available from normal commercial sources or from a prearranged source not subject to the same external event that may occur at the site, and to be placed in readiness at the plant within the initial 72-hour period, as needed. In the rationale portion of revised Section 2.2.10, EPRI states that the 72-hour period and the additional margin of extended passive system operation will provide a time margin for the repair and/or replacement of non-safety defense-in-depth equipment if such equipment is damaged by an internal or external event, and for organization of offsite assistance.

Revised Section 2.2.11 to Chapter 5 requires that the recovery process (e.g., from safe state to long-term maintenance of cold shutdown), including time and plant conditions and equipment to be utilized to achieve cold shutdown, will be defined by the plant designer as part of the severe accident management program specified in Chapter 1, Section 2, and Chapter 5, Section 2.4.3.4, of the passive Requirements Document. Passive engineered safety systems equipment is required to be designed and environmentally qualified for the extended period described in Section 2.2.10 of Chapter 5 of the passive Requirements Document, and until such time that recovery can be ensured. If passive safety system operation is terminated before the end of the recovery period (i.e., by operation of non-safety systems utilized in the recovery process), the capability to restore operation of the passive safety systems is required.

In the rationale portion of revised Section 2.2.11 of Chapter 5, EPRI states:

> It is important that safety systems be qualified for the entire postaccident period. The recovery operation may utilize the passive safety systems, onsite non-safety systems, and offsite support to the extent necessary to achieve and maintain cold shutdown. The ability to return to passive safety systems operation is important during this period of plant recovery in order to protect against the possibility of failure of the support of non-safety systems or of offsite support.

For events extending beyond 72 hours, EPRI, in its January 10, 1992, response on this issue, also described "typical" methods of removing decay heat; providing inventory makeup to passive safety system pools; providing cooling for long-term instrumentation, equipment, and habitability environments; and powering the safety loads. EPRI indicated that the specifics of these methods will be left to the plant designers. In its description EPRI generally described the operation of the boiling-water reactor (BWR) high-pressure isolation condenser and the pressurized-water reactor (PWR) high-pressure passive decay heat removal heat exchangers and the water pools in which they will be located. It indicated that the quantity of water in the BWR water pool will be sufficient for keeping the heat exchanger covered for about 72 hours because the water vapor will be released to the environment; in the case of the PWR, the water in the pool will be replenished by condensation of the evaporated steam on the inner surface of the containment and is, therefore, expected to last much longer than 72 hours. (In both designs, however, the natural circulation heat exchangers will not be able to cool the RCS temperature to cold-shutdown levels that are typically in the range of 200 °F). EPRI stated that both designs will be capable of continued functioning, requiring only simple actions after 72 hours to provide additional water

used to replace boiloff; this makeup to the passive safety pool(s) will be provided through a dedicated, seismic, safety-class pipe routed to outside the reactor building for use in emergency situations when normal makeup may not be available. In response to a separate staff question on replenishment of the passive decay heat removal pools, EPRI, in its January 10, 1992, response, stated that demineralized water tanker trucks from outside the region affected by external events could be one source of offsite water supply to the passive safety system pools. EPRI also stated that, as backup to the offsite sources, water may be obtained from the onsite fire water system, which will be provided with a diesel-driven pump designed to withstand the effects of an SSE and remain functional. EPRI identified requirements that it subsequently added to Chapter 5, Section 2.2.16, to ensure that provisions for the emergency water sources are provided. As stated above, the new requirements specify that a seismic Category I emergency water makeup connection be brought out to an accessible area and be of a design readily adaptable to the portable water supplies from offsite water sources (e.g., demineralized water tanker trucks). EPRI also requires that a connection be provided to the emergency water makeup piping from the fire protection system to function as a backup onsite water source. To clarify the role of the diesel-driven fire pump, EPRI revised Section 3.4.3.3 of Chapter 9 so that one of the sources of the fire water supply (including storage) and the diesel-driven fire pump (including suction line and pump discharge line to the fire water main) will be analyzed to withstand the effects of an SSE and remain functional. The rationale for the SSE requirement on the diesel-driven fire pump is to provide further assurance of its availability as a backup for safety pool makeup under emergency conditions.

Regarding "typical" methods of providing cooling for long-term instrumentation, equipment, and habitability environment, EPRI stated that cooling for instrumentation and equipment, other than for the control room, was not expected to be required on the basis of the environmental profile that will be specified for the appropriate accident qualification envelope. However, if necessary, hardened connections will be designed to provide support, such as outside air cooling. (Note: The staff interprets the term "hardened" as a term used to describe a system or feature that is designed to withstand an SSE and remain functional.) EPRI stated that these or similar methods could be used to ensure functionality of items located within the primary containment as well as other structures containing safety-class equipment and instrumentation. Control room continued habitability and cooling for the control room instruments and controls may be provided by pressurized bottled air for up to 72 hours. Following the 72-hour period, EPRI stated that such bottles will be easily replenished and/or dedicated connections will be provided in the design to allow for the use of external air supplies. Regarding "typical" methods of powering safety loads, EPRI stated that the safety loads are capable of accepting offsite supplied power generated by transportable units. This statement is also supported by requirements for such a capability in Chapter 11.

EPRI also addressed the staff's question regarding the use of non-safety ac electrical systems to maintain passive safety systems in a state of readiness preceding an event. EPRI stated that during normal plant operating conditions, non-safety electrical distribution systems will provide power to the various non-safety system loads and will be used to maintain batteries in a charged state while also providing the normal supply to the safety loads.

Similarly, EPRI stated, passive safety system pools will maintain inventory by means of non-safety system supplies, with appropriate levels, temperature, and such being the basis of the technical specifications. EPRI maintained that the use of the non-safety systems (electrical and mechanical) to support safety systems in maintaining states of readiness, as described above, is traditional in existing light water reactor designs and is continued as a design feature for passive plants as proven technology.

On the basis of EPRI's responses on this issue, the revisions EPRI has made to the passive Requirements Document, and the staff's review of the passive Requirements Document, the staff concludes that EPRI has required that provisions be made in the passive plant design to allow support from offsite equipment and backup from onsite facilities in order to extend the passive safety system capability well beyond 72 hours. The onsite facilities and the connections to utilize offsite support are required to be readily accessible and to withstand SSE conditions. The offsite equipment and resources needed to maintain the plant in a safe state are required to be readily available, not subject to the same external event that may occur at the site, and to be placed in readiness at the plant within the initial 72-hour period. EPRI has not specified any reliance on any part of the non-safety electrical systems in order to extend the use of the passive systems beyond 72 hours, nor does it appear that EPRI intends that such systems be relied on for this purpose. On this basis, the staff concludes that there is no need to require that the non-safety electrical systems (both non-safety and permanent non-safety) have any additional capabilities or qualifications (other than those already specified by EPRI) in order to support passive safety system use beyond 72 hours. If the plant designs do rely on any portion of the non-safety electrical systems to support passive safety system use beyond 72 hours, the staff will reevaluate this issue during its review of an individual application for FDA/DC. Also, as discussed in Section 2.2.1 of this chapter, the staff identified an issue concerning RTNSS in passive plant designs that addresses the role of the non-safety systems as backup to the passive safety systems for defense-in-depth purposes. In accordance with the process identified in SECY-94-084 for the RTNSS issue, additional requirements may be placed on the non-safety electrical systems commensurate with their risk significance identified by the plant designers under the RTNSS process.

Although EPRI has not identified any roles the non-safety electrical systems are required to play in extending passive safety system operation beyond 72 hours, it also has not identified any alternatives to the use of the non-safety systems to bring the passive plant to a cold shutdown condition. In revised Section 2.2.11 of Chapter 5, EPRI requires that the recovery process (e.g., from safe state to long-term maintenance of cold shutdown), including time and plant conditions and equipment to be used to achieve cold shutdown, be defined by the plant designer as part of the severe accident management program specified in Chapter 1, Section 2, and Chapter 5, Section 2.4.3.4, of the passive Requirements Document. The staff will evaluate the information in the severe accident management program referenced in Section 2.2.11 during its review of an individual application for FDA/DC. The review will be conducted to ensure that if the non-safety electrical systems are relied on to bring the plant to cold shutdown, sufficient provisions have been included and time will be available beyond 72 hours to repair and recover those systems if they are damaged by conditions surrounding or initiating an event.

2.2.3 Compliance With Intent of Regulations, Regulatory Guides, and Standard Review Plans in the Electrical Area

Section 4 in Appendix B to Chapter 1 of the passive Requirements Document addresses regulations, regulatory guides, and standard review plans that the passive ALWR may not specifically comply with, but which EPRI states the passive ALWR still meets the intent of. A number of the identified regulatory requirements and guidance are in the electrical area, and for many of those that address ac power, the analysis of passive ALWR compliance with intent is essentially that the passive safety systems meet the intent of the requirement and, therefore, ac-powered safety-related electrical systems are not required. The implication is that the specifics of the regulatory requirement or guidance will not be met. However, in other sections of the passive Requirements Document, requirements are specified that replace, at least to some extent, the requirements of the regulatory position. For example, in Section 4.20 of Appendix B to Chapter 1, with regard to Regulatory Guide (RG) 1.75, "Physical Independence of Electric Systems," EPRI states that the only required electric power is dc power that will meet the physical independence requirements of RG 1.75. However, Section 2.3.13 of Chapter 11 and Section 2.3.3.3 of Chapter 3 require that the two non-safety ac divisions be separated to the maximum extent practical. Because the various requirements pertaining to the regulatory issues are often scattered throughout the passive Requirements Document, it is difficult to determine to what extent the regulatory guidance will be met. As a result, the staff asked EPRI to amend the regulatory items in the electrical area addressed in Section 4 of Appendix B to Chapter 1 to identify which portions and to what extent the passive ALWR will comply with the specific recommendations of the regulatory guidance.

In the response dated January 10, 1992, EPRI replied that the intent of the "comply-with-intent" section of Chapter 1, Appendix B, was to give a high-level indication of regulations and guidance where the detailed design-specific analysis requested in the staff's question might be necessary. However, EPRI stated, with respect to electrical system considerations, the comply-with-intent items can be grouped and the expected treatment of these groups can be addressed. The electrical comply-with-intent items were then divided into three groups and EPRI addressed each group separately. In the first group, EPRI stated that the division between "applicable" and "not applicable" sections of the regulation and guidance was clear because some sections are simply not applicable to the non-safety ac power systems. In the second group, which consists of guidance directed at safety-related diesel generators, EPRI stated that the documents were "not applicable" in their entirety because the passive plants do not have safety-related diesel generators. For the third group of comply-with-intent issues, which concern the effect of loss of ac power on the public health and safety, EPRI concluded that the demonstration of compliance should be expected to be much simpler than that traditionally provided because of the substantially lowered safety significance of the loss-of-ac-power events.

The staff did not necessarily agree with EPRI's conclusions regarding the expected treatment of the three groups of electrical items as they pertain to all the regulations and guidance identified in each group. In its DSER for Chapter 11, the staff stated that resolution of its concerns relative to the regulatory treatment of non-safety systems (RTNSS) referenced in Section 2.2.1 of Chapter 11, as well as the potential reliance on non-safety ac systems

during shutdown modes, might require that at least some of the enhancements prescribed in those guides and regulations be made to the non-safety ac electrical power systems. The staff, however, agreed with EPRI that a detailed design-specific analysis would likely be necessary to adequately address many of the guidance documents and regulations. The staff addressed some of the guidance and regulations individually in the DSER for Chapter 11. The staff stated that it would evaluate the remaining items during its review of an individual application for FDA/DC.

Subsequently, a process was defined for resolving the RTNSS issue as noted by the staff in Section 2.2.1 of this chapter. This process can be used as a screening tool to help the staff determine which of the subject regulatory requirements dealing with ac power system features may need to be pursued for the passive designs. To aid in resolving this issue, the staff will require that the non-safety standby power sources and ac distribution systems be addressed during the design-specific RTNSS process. If the non-safety standby power sources or non-safety ac distribution systems are determined to be risk significant under this process, the staff will consider the option of further pursuing the applicable regulatory requirements. It will, therefore, address this issue during its review of an individual application for FDA/DC.

2.2.4 Minimization of Class 1E Components

Section 2.3.8 of Chapter 11 originally required that the number of Class 1E components be kept to a minimum. Equipment or systems that were not essential for emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal, or were not otherwise essential in preventing significant release of radioactive material to the environment, would not be designed Class 1E unless they constituted auxiliary equipment clearly required for the operation of Class 1E components.

An identical requirement was specified for the evolutionary ALWR in Chapter 11, Volume II of the EPRI Requirements Document. Because EPRI had replied to staff concerns on this issue in that review, the staff asked if EPRI had any additional information on this subject relative to the passive ALWR. In a January 10, 1992, letter, EPRI replied that no additional information was available at that time on this subject. The two paragraphs that follow, therefore, document the staff's review and conclusions on this matter for the evolutionary ALWR, which is also the staff's position for the passive ALWR.

In a letter dated April 10, 1990, the staff expressed its concern about the requirement that the number of Class 1E components be kept to a minimum because it allowed non-safety concerns (minimizing qualification, surveillance, and maintenance) to potentially affect safety improvements and could be misinterpreted or abused, resulting in the downgrading of components that had formerly been designed to Class 1E requirements. By letter dated July 23, 1990, EPRI responded that it required that the safety systems be designed for simplicity of operation, surveillance, and maintenance so as to optimize their reliability and to keep cost down. EPRI indicated that higher levels of safety than those of existing plants will be achieved by specifying other safety improvement requirements; therefore, the staff's concern that this approach could lead to a downgrading of safety or performance, was unfounded. EPRI also indicated it believes that it is necessary to have definitive requirements in this area to provide a basis for standardization and to avoid

the large number of "custom" designs. Finally, with regard to the staff's position that some level of qualification, surveillance, and maintenance requirements should be specified for some non-safety equipment commensurate with its importance to safety, EPRI stated that the staff had raised a similar concern in the DSER for Chapter 5 of the evolutionary Requirements Document and that it would address the appropriate level of qualification for specific equipment and systems in that chapter. EPRI stated that Chapter 11 will be revised as appropriate to be consistent with Chapter 5.

The staff disagreed with EPRI that the subject requirement was "definitive" and would lead to a more standardized use of Class 1E and non-Class 1E equipment. Because the requirement was open to interpretation regarding what was "essential" and what constituted "auxiliary equipment required for the operation of Class 1E components," the staff could not determine to which systems or components the requirement applied. For example, the staff could not determine if this applied to the categorization of electrical protective overcurrent relaying and electrical monitoring instrumentation. Also, broad safety goals do not provide adequate assurance of well-designed and well-specified systems at this level of detail. The staff concluded that the resolution of the level-of-qualification issue in Chapter 5 of the passive Requirements Document would not identify electrical equipment or components down to the level of detail necessary for the staff's review. In the DSER for Chapter 11, the staff stated that it would, therefore, evaluate how this requirement for minimization of Class 1E components has been applied during its review of an individual application for FDA/DC.

By letter dated September 29, 1992, EPRI responded to this issue, stating that the purpose of this requirement is not to reduce the number of Class 1E components so as to downgrade Class 1E structures, systems and components (SSCs) but to require the designers to develop the necessary analyses to comply with the regulations without unwarranted conservatism that might result in an unjustified high number of Class 1E components. To avoid any misinterpretation EPRI, therefore, provided the following modification to Chapter 11, Section 2.3.8, in Revision 4 of the passive Requirements Document:

> Requirement:
>
> The number of Class 1E components shall be optimized so as to comply with the applicable regulations while avoiding unwarranted conservatism that lead to upgrading the plant components to Class 1E requirements without justification. The plant designers shall make detailed system analyses to determine what systems, equipments, components, and parts are essential to perform or support the safety-related functions and shall assign the classification and qualification requirements accordingly.
>
> Rationale:
>
> Many times in the past, during the design phases, overly conservative assumptions were made about the scope of application of Class 1E requirements to the systems of the plant. Although this was done to avoid costly detailed analyses, it, however, resulted in more costly O&M [operation and maintenance (O&M)] requirements along the life cycle of those components. We can see examples of this in

the [quality classification lists], [equipment qualification], seismic qualification, and replacement parts qualification programs. This requirement will prevent that situation without reducing the number of Class 1E components so as to downgrade the safety-related SSC.

The above changes indicate that it is not EPRI's intent to pursue the downgrading of electrical component qualification from that required by existing applicable regulations. The staff will evaluate the qualification of electrical SSCs during its review of an individual application for FDA/DC.

2.2.5 Non-Safety Power Supply Circuits Connected to Input Terminals of Class 1E Distribution Systems

Section 2.3.9 of Chapter 11 requires that the electric power distribution systems be designed so that non-safety circuits will not be connected to safety circuits or power sources. It notes, however, that this requirement does not apply to non-safety power supply circuits that will be connected to the input terminals of the Class 1E distribution systems. The staff indicated to EPRI that this note should be supplemented with the requirement that these non-Class 1E/Class 1E interfaces include Class 1E power regulation devices or Class 1E protective devices to isolate the Class 1E loads from out-of-tolerance power outputs from the non-Class 1E sources that exceed the rated values of the Class 1E loads.

In the January 10, 1992, letter, EPRI responded that a new requirement would be included in Section 2.7.6 of Chapter 11 of both the evolutionary and the passive Requirements Document.

In the DSER for Chapter 11, the staff concluded that the proposed change was consistent with its recommendation and was, therefore, acceptable. This was identified as a confirmatory issue.

After the staff's DSER was issued, in its "Branch Review Book" for the Electrical Engineering Branch, forwarded by letter dated September 29, 1992, EPRI gave a position on this issue that was different from that in its January 10, 1992, letter. It indicated that, on the basis of its understanding of the NRC position, isolation devices based on degraded voltage conditions were being required. EPRI stated that in the passive plants the only expected Class 1E loads that will be powered from non-Class 1E power supplies will be the Class 1E battery chargers and the vital ac regulating transformers. It indicated that both types of equipment will be designed to tolerate voltage oscillations and to filter them out so they do not affect the downstream Class 1E loads. Under accident conditions, EPRI stated, the main ac system of the plant could generate voltage oscillations, caused by, for example, load sequencing. It did not, however, consider it adequate to disconnect the Class 1E chargers/regulating transformers from their normal power source under these conditions because this would imply the loss of an important defense-in-depth capability of the electric power system. EPRI proposed, instead, isolation devices in accordance with IEEE 384 and RG 1.75 and modified Section 2.7.6 of Chapter 11 of the passive Requirements Document as follows:

Requirement:

Where non-safety power supply circuits are available to supply power to safety systems, isolation devices, as described in IEEE 384, shall be provided to isolate the safety systems from their non-safety sources so that in case of a DBE [design-basis event], postulated electrical faults in the non-Class 1E source will not damage the Class 1E loads so they lose their capability to perform their intended safety function. These isolation devices shall be qualified as Class 1E.

Rationale:

To comply with IEEE 384 requirements endorsed by RG 1.75.

In a letter dated April 14, 1993, EPRI provided additional information on this issue. It stated that Class 1E isolation in accordance with IEEE 384 is needed to prevent the Class 1E batteries from discharging through the battery chargers into a fault on the non-Class 1E ac side of the chargers. With regard to degraded voltage vulnerabilities, EPRI stated that the passive designs meet the intent of the staff position on degraded voltage (Branch Technical Position PSB-1 in the Standard Review Plan) because they do not rely on ac power for their safety-related missions. EPRI stated that the battery chargers by themselves are not required to perform safety functions other than the above-described isolation function, and a degraded voltage condition disabled a battery charger, the battery would provide Class 1E power to supply the corresponding safety loads for 72 hours. EPRI stated that 72 hours was long enough to repair the charger or to provide charging current through the spare charger.

The staff agrees that isolation capability should be provided to prevent the Class 1E batteries from discharging into faults on the ac side of the battery chargers. It requires this capability at existing plants as well, to isolate the four-channel instrumentation dc power supplies from the two-division ac systems that normally power them through the battery chargers. This isolation capability is normally provided by the chargers themselves by way of the unidirectional electronic devices such as silicon controlled rectifiers (SCRs) and diodes that are used to convert the ac current to a dc current.

The staff also agrees that the battery chargers and the regulating transformers should be designed to tolerate voltage perturbations (and in the case of the battery chargers, frequency perturbations as well) and filter them out so they do not affect the downstream loads. However, if the Class 1E battery chargers and regulating transformers can be damaged by abnormal voltage or frequency from the non-Class 1E power sources, power regulation devices or protective devices should be inserted between the source and its loads to prevent that damage. Spurious separation of the chargers or transformers from their sources as a result of normal transients such as load sequencing can be prevented by the appropriate use of time delays in the protective devices.

Preventing damage to the chargers and transformers would allow them to be available after 72 hours when portable ac power sources would be brought on site to recharge the batteries and provide dc power. The chargers are required to be available at this time because they are needed to convert the

portable ac power to dc power to recharge the batteries and supply the Class 1E dc loads. The passive Requirements Document provides only for connecting portable ac power to the plant. It does not provide for connecting portable dc power directly to the Class 1E dc system, nor does it provide for repairing the Class 1E battery chargers or regulating transformers. The spare battery charger cannot be relied on to provide the ac to dc conversion capability because it must be energized from the ac system, at least occasionally, to keep the spare battery in a ready state of charge. It would, therefore, be vulnerable to the same ac power system abnormalities as the normal chargers. Also, the spare battery charger may not be sufficient to power all the Class 1E loads required after 72 hours if both 2-hour and 72-hour safety loads were needed at that time. EPRI has expressed concern that inserting protective devices would unnecessarily increase the complexity of the dc systems and the chances of losing all the ac sources to the charges, if these were designed with built-in features that isolate them or prevent their damage in case of abnormal voltage/frequency. The staff agrees that external protective devices can be exempted if the Class 1E loads include built-in features that either isolate or prevent damage to the load in case of abnormal voltage and/or frequency. The staff will pursue this issue with the passive plant designers during its reviews of individual applications for FDA/DC.

2.2.6 Need for the Non-Safety AC Power Systems During Shutdown Conditions

Section 2.3.12 of Chapter 11 specifies that the electrical systems - safety and non-safety - be arranged so that safe shutdown can be achieved with all equipment in any one fire area rendered inoperable by fire, smoke, hot gases, or fire suppressant, without possible recovery. "Safe shutdown" for the passive ALWR is defined in Section 2.5.6 of Appendix B to Chapter 1 as reaching a shutdown condition of about 420 °F in 36 hours. This can be accomplished by the passive systems alone. The Section 2.3.12 requirement, therefore, could be met by only protecting the safety-related dc and associated vital ac systems from the effects of fire, smoke, hot gases, or fire suppressant. The passive systems, however, cannot bring the plant to a cold-shutdown condition nor can they maintain it in a cold-shutdown condition. If the non-safety ac electrical systems are damaged by fire, smoke, hot gases, or fire suppressant, it may not be possible to recover them for a significant period.

Section 2.3.13 of Chapter 11 requires that non-safety equipment and circuits pertaining to one of the two non-safety divisions be physically separated to the maximum extent practical from equipment and circuits pertaining to the other non-safety division. The lack of specific separation criteria here could make the non-safety systems vulnerable to failure that could not be repaired for a significant period of time.

Section 2.6.1.5 of Chapter 11 specifies that non-Class 1E equipment and its support structures will be designed to meet the requirements of seismic Category II if they are located in the vicinity of Class 1E equipment or support structures. Seismic Category II, as defined in Section 4.3 of Chapter 1, does not require that the equipment remain functional during or following an earthquake, but only that it not structurally fail or interact in such a manner as to degrade the functioning of seismic Category I structures.

The non-Class 1E equipment, therefore, has no requirement that it remain functional following an earthquake. The lack of seismic capability might make the non-safety systems vulnerable to failures caused by a seismic event that could not be repaired for a significant period.

Such features of the non-safety-related ac electrical power systems could render these systems inoperable for significant periods. EPRI has addressed the staff concerns identified in Section 2.2.2 of this chapter relative to loss of the non-safety ac power systems during power operation. In its response to that issue, EPRI committed to a revision in the passive Requirements Document that would require the designer of the passive plant to define the plant recovery process from safe state to long-term maintenance of cold shutdown, including time and plant conditions and equipment to be utilized to achieve cold shutdown, as part of the severe accident management program. During shutdown at existing plants, however, the staff finds that loss of ac power during some modes, such as midloop operation in PWRs, can only be tolerated for less than an hour or two. The staff, therefore, stated in the DSER for Chapter 11 that it was concerned that in the passive plant designs, during shutdown when the passive safety systems might not be operable, the severe accident management program support equipment might not be available soon enough to prevent fuel damage following a loss of the non-safety ac power system (a portion of which is necessary to support decay heat removal). EPRI was, therefore, directed to address this issue by identifying how long a loss of non-safety ac power could be tolerated during shutdown in a passive plant design and by identifying the provisions that would be made to respond to this contingency within that period. The staff identified this as an open issue.

In Revision 4 to Chapter 5 of the passive Requirements Document, EPRI added Sections 2.2.17 and 2.2.18, which address shutdown issues. Section 2.2.17 requires that the plant designer evaluate a number of items for outage-specific conditions to ensure that the ALWR safety goals can be met. One of these items is an inventory of water sufficient to provide makeup for losses of reactor coolant system (RCS) volume for 72-hours. Also, Section 2.2.18 requires that valves capable of draining the RCS or the vessel be capable of being closed without reliance on ac power. This is to prevent the possibility of continuous draining if a station blackout should occur during shutdown or refueling modes. These provisions indicate that EPRI's intent is to at least minimize the reliance on non-safety ac systems during shutdown modes so that sufficient time (72-hours) would be available to enlist the use of the offsite support equipment if necessary. It is not clear from some of the other provisions in Section 2.2.17, however, whether ac power would or would not be needed during the 72-hour period.

The steps identified in SECY-94-084, however, specifically require that shutdown modes be included in the PRAs used to resolve the RTNSS issue. The non-safety ac distribution systems and standby ac power sources will, therefore, be addressed in the design-specific shutdown-focused PRAs. These PRAs are required for shutdown modes to determine the importance of non-safety systems during those modes. The staff will evaluate the results of those PRAs during its review of an individual, application for FDA/DC. It will pursue this issue further if the results demonstrate that the Commission's safety goals and regulatory requirements cannot be met without reliance on the non-safety ac distribution systems or standby ac power sources.

2.2.7 Use of Revisions to IEEE Standards Not Yet Endorsed by the Staff

Section 2.6.1.1 of Chapter 11 specifies that its requirements satisfy the latest revision of an IEEE standard, as modified by applicable regulations. This type of instruction is found throughout Chapter 11 whenever IEEE standards are referenced. The problem is that the NRC has often not yet endorsed the latest revisions of IEEE standards. This was also a problem in Chapter 11 of the evolutionary Requirements Document. The staff concludes that the latest revision of the IEEE standard could be referenced, provided that during the FDA/DC process, the plant designers identify all changes from, and additions to, the last version of the IEEE standard that was endorsed by an RG. The staff will then evaluate the changes and additions for acceptability during its review of an individual application for FDA/DC.

2.2.8 Minimum Starting Voltages for Valve Actuator Motors

Section 2.6.2.3 of Chapter 11 originally required that Class 1E valve actuator motors be capable of accelerating their loads as required with a starting voltage at the motor as low as 80 percent of rated motor voltage. This same requirement was specified in the evolutionary Requirements Document. In the passive ALWR design, however, these Class 1E valve actuator motors will use dc power as compared with ac power for the evolutionary ALWR. In general, dc systems are typically designed for operation down to a steady-state battery final voltage of 105 V (1.75 V/cell), which is 84 percent of the nominal system voltage of 125 V. If the motors are only sized for a starting capability of 80 percent, this leaves little allowance for voltage drop due to motor starting inrush currents. Therefore, the staff concluded that EPRI should specify that motors be required to start with a minimum voltage lower than 80 percent.

EPRI responded as follows in its January 10, 1992, letter:

> A requirement for sizing the valve actuator motors based on a standardized starting voltage value lower than 80 percent is undesirable as it would require oversizing not only all motors but also the operators and valves. The result would be unwanted size, weight, and cost increases which may be unwarranted in many cases. A starting voltage lower than 80 percent should only be required if necessary following design optimization of the power distribution circuits with regard to voltage drop. The number of dc valves is typically sufficiently small to permit detailed evaluations of the possible design tradeoffs. In order to ensure that all functional requirements for the safety systems are met, Section 2.6.2.3 of Chapter 11 will be revised as follows:
>
> Requirement 2.6.2.3:
>
> Class 1E valve actuator motors shall be capable of accelerating their loads as required with a starting voltage at the motor as low as 80 percent of rated motor voltage. The adequacy of the 80-percent starting voltage rating shall be demonstrated by calculations for each valve or a lower starting voltage shall be specified as appropriate based on the calculation results.

Rationale:

This requirement is intended to ensure that adequate margin for source and circuit voltage drops is provided to meet all functional requirements of the safety systems without undue oversizing of the valve operators. A starting voltage lower than 80 percent should only be required if necessary following design optimization of the power distribution circuits with regard to voltage drop.

In the DSER for Chapter 11, the staff concluded that these revisions satisfied its concerns that the minimum starting voltages specified for valve actuator motors would be sufficient to ensure starting of the motors under minimum dc distribution system supplied voltages. This was identified as a confirmatory issue. The staff has verified that these changes were made in Revision 3; therefore, this confirmatory issue is closed.

2.2.9 Integrity of Electrical Cable Penetration Seals

Section 2.6.4 of Chapter 11 requires plant designers to ensure that seals at locations where cables penetrate fire barriers will remain effective should cable trays collapse from the effects of fire. Chapter 10 of the passive Requirements Document references the Chapter 11 design criteria so that it is clear that penetrations of barriers for instrumentation and control cables must meet the same requirements. The staff concludes that EPRI's commitment meets the staff's enhanced fire protection criteria in Chapter 9 of this report and SECY-90-016 and is, therefore, acceptable.

2.2.10 Integrity of Bus Duct Penetrations

In its request for additional information dated August 16, 1991, the staff raised the issue about possible locations where bus ducts would penetrate fire barriers and asked EPRI to clarify how it proposed to design such penetrations to satisfy the 3-hour-fire rating criteria should such penetrations be allowed in the design criteria.

In its letter dated January 10, 1992, EPRI responded that only the isophase bus is expected to require air cooling in the design of a passive ALWR and that it will pass through the turbine building wall and connect to an oil-filled transformer located at least 50 feet from the building. EPRI also stated that although it does not expect that a 3-hour-fire rating will be required for this bus duct penetration, the fire hazards analysis required by Section 3.3.2.1 of Chapter 9 will assess the adequacy of the turbine building wall.

The staff concludes that EPRI's response meets the staff's enhanced fire protection criteria in Chapter 9 of this report and SECY-93-087 and is, therefore, acceptable.

3 OFFSITE POWER SYSTEM

3.1 Functional Description

Section 3.1 of Chapter 11 of the passive Requirements Document states that the offsite power system will include the set of electrical circuits and associated equipment that will be used to interconnect the offsite transmission system, the main generator of the plant, and the onsite electric power distribution systems. It will include the plant switchyard or remote station, the main step-up transformers, the unit auxiliary transformers, the high-voltage tie lines, and the isolated phase buses with their associated auxiliary systems, including protective relays and local instrumentation and controls.

In general, Section 3 of Chapter 11 calls for the offsite power circuit to use the main generator circuit in the backfeed direction from the switching station (plant switchyard or remote station) through the main step-up transformer to the unit auxiliary transformers. EPRI refers to this circuit as the "main offsite power circuit," and it will be the normal source of power for all plant loads (safety-grade, non-safety-grade, and permanent non-safety-grade) during all modes of operation (operating and shutdown). A low-voltage generator circuit breaker will be provided to isolate the main generator from this circuit when the generator is inoperative.

EPRI has not specified a requirement for a second offsite power circuit; however, it has specified that the design of the plant electric power systems will permit each division of permanent non-safety-grade loads to be provided with an additional maintenance power supply circuit for use during plant outages.

3.2 Evaluation

3.2.1 Lack of a Second Offsite Power Supply Circuit

Section 1.5.2 of Chapter 11 states that the ALWR passive plant design does not require a reserve offsite power supply circuit since such a circuit would not provide any significant additional safety or availability benefit. Section 2.3.16 of Chapter 11 provided the specific requirement in this area. It originally stated that the design of the electric power systems would permit each division of permanent non-safety-grade loads to be provided, at the option of the plant owner, with an additional maintenance power supply circuit for use during plant outages. However, as stated in Section 1.5.2, there were no specific requirements that a second offsite power supply circuit be provided. As a result, the staff asked EPRI to address the lack of a second offsite power supply circuit.

In a letter dated January 10, 1992, EPRI gave the following response:

> As indicated in the response to Question 435.1, passive safety systems, which are independent of ac power for operation will provide decay heat removal and maintain the plant in a safe shutdown condition for 72 hours following a licensing design bases event. This capability will be maintained for an extended period of time by

providing readily available offsite support resources. In contrast, existing plants rely on ac power to operate their safety systems and cannot maintain the plant in a safe shutdown condition without ac power for more than a few hours. It follows that while a station blackout (SBO) event results in a degradation of the safety functions at an existing plant it has little impact on the capability of the safety systems of the ALWR passive plant.

The main purpose of requiring passive safety systems is to eliminate the failure modes associated with active systems, including the power supply sources. In order to ensure that the benefits of passive systems are realized, a requirement for an overall core damage frequency of 1.0E-5 per reactor-year was set for the ALWR passive plant. This requirement results in a level of plant safety substantially higher than for existing plants. SBO is, therefore, not a safety issue for the passive plant and no requirement should be imposed on SBO frequency for this plant.

SBO is, however, considered an investment protection issue. This issue is satisfactorily addressed in the ALWR passive plant design by requiring two onsite standby sources of ac power for important plant loads in addition to a reliable normal power supply. In order to further improve the reliability of ac power supply, particularly, during plant shutdown, the additional maintenance power supply circuit now described as optional in Section 2.3.16 of Chapter 11, Volume III will be required as a standard design feature for the ALWR passive plant. Accordingly, Section 2.3.16 of Chapter 11, Volume III will be revised as follows:

Requirement:

The design of the plant electric power systems shall provide for a separate power supply circuit to the plant permanent non-safety loads for use in the event of an extended unavailability of the normal power supply, e.g., during plant outages. This additional circuit shall consist, for example, of a tie to another generating unit at the same site or, in the case of a single-unit sites, of a tie to the plant switchyard or a separate offsite source. As a minimum, this circuit shall have sufficient capacity to supply power to the permanent non-safety loads provided to maintain cold shutdown and to normal maintenance loads.

Rationale:

This requirement will ensure that adequate power supply will be maintained (either from another offsite source at the same site or from off site) at all times during plant shutdown modes when major maintenance is required on one of the onsite power sources or on the normal offsite circuit.

Section 4.2.3 of Chapter 11, Volume III will also be revised to eliminate the conditional statement in the first sentence. This first sentence will now read as follows:

The transfer of the plant's permanent non-safety loads to the maintenance circuit required in Paragraph 2.3.16 shall be strictly manual.

In the DSER for Chapter 11, the staff concluded that these changes to the passive Requirements Document proposed by EPRI were acceptable insofar as they applied to the need for a second offsite power circuit during plant shutdown conditions. This was with the understanding that the "tie to another generating unit at the same site" specified in the requirement provides access to a second offsite circuit that is separate from the normal offsite circuit, and that the second offsite circuit meets the criteria required by General Design Criterion (GDC) 5, "Sharing of Structures, Systems and Components," in Appendix A to 10 CFR Part 50.

It was not clear from the proposed requirement, however, what the availability of the second circuit would be during power operations. The staff stated in the DSER that lack of a second offsite power source during power operations would result in more frequent total loss of offsite power events (which could lead to SBO) and longer times for recovering offsite power.

The staff further concluded that with regard to SBO, even with a second offsite circuit available, the frequency of SBO would be potentially increased at the passive ALWRs because of the lower availability required of the standby power sources (whether diesel generators or combustion turbine-generators) and the more relaxed design requirements specified for the standby power sources, such as the lack of two air start systems and reduced capability to start and operate under wide ambient air temperature extremes. In the staff's opinion, the frequency of SBO should not be further increased by the lack of the availability of a second offsite power circuit during power operations.

The staff additionally concluded that the lack of the second offsite power circuit during power operations could also result in longer times for recovering offsite power following SBO events and total loss of offsite power events. Failure (faults, short circuits) of the one offsite power circuit in the passive plant designs would result in a total loss of offsite power and a plant trip. Because the only available offsite circuit would be damaged in these events, recovery of offsite power would be prolonged and reliance would have to be placed on extended operation of the standby power sources, or reliance on the passive safety systems alone would be required if the event resulted in an SBO.

EPRI indicated in its responses that, although an SBO event results in a degradation of the safety functions at an existing plant, it has little impact on the capability of the safety systems of the ALWR passive plant. In the DSER, the staff agreed that an SBO might not degrade the passive safety systems, but it stated that it did challenge those safety systems, and because of the concern it identified in Section 2.2.1 of this chapter relative to the uncertainties in the performance of the passive features, it believed those challenges should be kept to a minimum.

EPRI stated that, to ensure that the benefits of passive systems are realized, a requirement for an overall core damage frequency of 1.0E-5 per reactor-year was set for the ALWR passive plant. EPRI indicated that this requirement results in a level of plant safety substantially higher than for existing

plants, and SBO is, therefore, not a safety issue for the passive plant and no requirement should be imposed on SBO frequency. The staff disagreed. Top-level core damage frequency goals should not be used as a basis to eliminate major support features such as a second offsite power circuit when the probabilistic risk assessment performed to substantiate adherence to that goal may contain substantial uncertainties for a passive ALWR because of its unique nature. For example, failure modes unique to the passive plants may not be adequately treated, and failure rates used for passive plant features may be more representative of an evolutionary plant than a passive plant. The staff provided specific examples where inappropriate evolutionary plant-type data were proposed to model passive plant systems. These data are addressed in detail in Sections 3.2.3 and 5.2.4 of this chapter, which follow.

In the DSER for Chapter 11, the staff concluded that because lack of a second offsite power circuit during power operations would result in more frequent total loss of offsite power events, longer times for recovering offsite power, and more frequent SBO events, a second offsite power circuit should be available to the second tier (permanent non-safety-grade load buses) in passive ALWRs during power operations. The rationale for this position was to ensure that the full benefit of reliability gains achieved by the use of passive systems was realized rather than used as a basis to allow increased SBO frequencies and longer offsite power recoveries. This is consistent with the principle of defense-in-depth and the philosophy that safety systems should not be unnecessarily challenged. The staff stated that an increased safety margin for loss-of-ac-power events is important to compensate for any new or unfamiliar passive plant failure mechanisms. This is consistent with the evolutionary plant policy that calls for an alternate ac (AAC) power source (as defined in 10 CFR 50.2) to cope with an SBO. The addition of such an AAC source provides the evolutionary plant with additional safety margin over existing operating plants. Similarly, the staff reasoned that the increased coping capability of the passive plant should be used to provided increased safety margin over existing operating plant capabilities. This was identified as an open issue.

In its letter dated September 29, 1992, EPRI responded to this issue, stating that the ALWR program considers that the capability of the passive systems to manage an SBO together with the low probability of such events makes the alternate source an unnecessary feature for the passive plants. However, since this issue is related to the RTNSS issue, EPRI stated that the industry position will be established according to the resolution of that generic issue.

The staff agrees that this issue is related to the RTNSS issue and could be addressed under that issue. As discussed in Section 2.2.1 of this chapter of this report, the resolution of the RTNSS issue calls for the establishment of the importance of the non-safety SSCs in a particular passive design by means of a design risk assessment in order to determine the need for regulatory treatment of those SSCs. If this approach is to be used on the offsite system, it is imperative that the offsite system model used in the risk analysis accurately represents the offsite system in the passive designs. In Section 3.2.3, which follows, the staff discusses a concern it has had in this regard in the passive Requirements Document. The staff intends to pursue that concern in its reviews of individual application for FDA/DC. Assuming satisfactory resolution of the concern, the staff concludes that the steps

identified in SECY-94-084 that to resolve the RTNSS issue for individual passive plant designs can also be used to resolve this issue. During its reviews of individual applications for FDA/DC, therefore, the staff will evaluate the results of the designers' analysis in this regard and will pursue additional regulatory requirements on the offsite system if the lack of a normally available second offsite circuit is found to be risk significant or is relied on to meet the availability missions.

3.2.2 Lack of Second Power Source for Non-Safety-Grade First-Tier Loads

With regard to the need for a second power source to the upper level (first-tier) non-safety-grade buses that was a policy issue pertaining to the evolutionary plants, the staff stated in its DSER that it had not determined the applicability of this issue to the passive designs. Because this issue is enveloped for the passive designs under the RTNSS issue discussed by the staff in Section 2,2.1 of the DSER, the staff stated that following final resolution of the RTNSS issue, it would report on its resolution for the passive ALWR.

As noted in Section 2.2.1 of this chapter, a process has been described in SECY-94-084 for resolving the RTNSS issue for individual passive plant designs. Also in SECY-94-084, the staff recommended that the electrical distribution issue on passive ALWR design be resolved by evaluating the ac power systems features using the RTNSS process. In its SRM dated June 30, 1994, the Commission approved the staff's recommendation on RTNSS as well as the recommendation that the RTNSS process be used to resolve the ac power systems issues. Therefore, the staff will require that the lack of a second power source to the non-safety-grade first-tier loads be addressed in the design-specific RTNSS process. If the lack of the second power source is determined to be risk significant under this process, the staff will consider the option of pursuing this issue further. It will, therefore, address this subject during its review of an individual application for FDA/DC.

3.2.3 Probabilities for Nonrecovery of Offsite Power as a Function of Time After Loss

Figure A1-1 and Table A1-2 in Annex A of Appendix A to Chapter 1 (PRA Key Assumption and Groundrules) establish the probabilities for nonrecovery of offsite power as a function of time after loss, to be used in the passive plant probabilistic risk assessments. According to the information in Section A1.2 in Annex A, these nonrecovery probabilities were developed by compiling the restoration times from 54 events identified in NSAC-147 for existing plants. Existing plants, however, typically have two offsite circuits, while the EPRI passive ALWR will have only one. The expectation for the passive plant would, therefore, be that its offsite power restoration times would typically be longer than those of existing plants. It does not appear, however, that this has been considered in establishing the nonrecovery probabilities identified in Figure A1-1 and Table A1-2. In fact, the nonrecovery probabilities identified there for the passive plant, which will have one offsite circuit, are actually better (more quickly recoverable) than those identified for the evolutionary plant in Figure A2-1, Annex A of Appendix A to Chapter 1 of the evolutionary Requirements Document - even though the evolutionary plant will have two offsite circuits.

In the DSER for Chapter 11, the staff stated that the above discrepancy would not be as great should the passive plant have a second offsite power circuit to the permanent non-safety-grade loads during power operations; but it would still exist.

Plant designers should, therefore, resolve this discrepancy and should ensure that their PRA models are consistent with their design and the Chapter 11 passive Requirements Document requirements. The staff will pursue this issue during its review of an individual application for FDA/DC.

3.2.4 Grid-Based Off-Normal-Frequency Conditions

Section 3.4.4.1 of Chapter 1 requires that the ALWR passive plant be designed to continue supplying the grid under off-normal-frequency conditions for a significant time. It states that during frequency excursions, it will be permissible for the voltage at the terminals of auxiliary equipment to vary from 100 percent to 90 percent of rated voltage. The rationale for the requirement is to allow the plant to contribute to maintaining grid integrity in the face of an undercapacity or rapid loss of load on the grid, or, in the event the grid cannot be maintained, to allow it to support a transition to islanding operation. The staff asked EPRI if the effect on the operation of equipment that uses ac power had been considered in the preparation of this requirement. For example, National Electrical Manufacturers Association Standard MG 1 specifies that ac motors will have the capability to operate successfully with a maximum combined variation in voltage and frequency of 10 percent (sum of the absolute values) of the rated values. The Section 3.4.4.1 requirement would allow plant excursions outside this range. The staff also asked EPRI if the effect on the design of the bus voltage monitoring and protection schemes specified in Section 4.5.3 of Chapter 11 had been considered. The voltage excursions permitted could result in actuation of this protection.

In its letter dated January 10, 1992, EPRI provided the following response:

> The implications of Section 3.4.4.1 of Chapter 1 regarding operating conditions of the plant loads have not been specifically analyzed, since this can only be done by means of detailed design calculations which cannot be performed at this stage. Section 3.2.4 of Chapter 11 is, however, intended to ensure that these analyses are performed and that the operating conditions of the plant loads are evaluated and found acceptable for all relevant grid conditions. The requirement of Section 3.4.4.1 is not expected to be a source of problems for the plant designer since the effects of voltage and frequency excursions on the motors would be mostly thermal and would only be of concern if these excursions exceeded reasonable values and were maintained for relatively long periods of time (relative to the motor's life expectancy). The effect on the design of the bus voltage protection schemes will be addressed as part of the detailed design of the protection systems.

The staff agrees with EPRI that the specified requirement will not likely be a source of problems for the plant designer relative to the effects on motors. However, other equipment, such as instrumentation and control equipment, or solid-state power supplies may be more of a problem. In either case, the

staff will evaluate the results of the analyses performed by the plant designer during its review of an individual application for FDA/DC to ensure that the operating conditions of all plant loads have been evaluated and found acceptable for all relevant grid conditions. The staff will also ensure, during its design-specific reviews, that the effect on the design of the bus voltage protection schemes has been addressed as part of the detailed design of the protection systems.

3.2.5 Availability of Second Offsite Power Circuits

In Section 1.5.2 of Chapter 11, EPRI states that the ALWR passive plant design will not require a reserve offsite power supply circuit, since such a circuit would not provide any significant additional safety or availability benefit. This view that a second offsite circuit would not provide any significant additional safety or availability benefit in a passive plant may extend from the use of passive safety systems that do not rely on ac power, but the staff is concerned that it also may result from the belief that a second offsite circuit has a high conditional unavailability when the normal offsite power circuit is lost. This may be the result of work done previously in establishing the loss of normal offsite power frequency for the evolutionary ALWR documented in Appendix A to Chapter 1 of the evolutionary Requirements Document.

In establishing the loss of normal offsite power frequency, Revision 1 to the evolutionary Requirements Document identifies 45 events taken from NSAC-147 for 1975 through 1989, corresponding to loss of normal offsite power. Of the 45 events, it states that the failures that occurred would have precluded the use of full-load rejection in 11 cases. By comparison, with regard to a policy issue on a second source of power to the evolutionary ALWR non-safety-grade loads, the staff identified 12 loss-of-normal-offsite-power events at U.S. plants in just 1988 and 1989 that would have precluded the use of full-load rejection. The failures identified were main step-up transformer faults, unit auxiliary transformer faults, and isolated phase bus faults. An additional three loss-of-normal-offsite-power events due to the opening of the main generator switchyard breakers were also identified in 1988 and 1989. Of the 15 total loss-of-normal-offsite-power events identified by the staff in 1988 and 1989, only one was identified in NASC-147. Also, in only 2 of the 15 events did the staff's data show that the alternate offsite power source failed to provide power to non-safety-grade reactor coolant pump buses. This gives a conditional unavailability of 0.13 for the alternate offsite power sources as compared to 0.8 or greater given by EPRI to the Advisory Committee on Reactor Safeguards in an April 12, 1991, meeting, and 0.36 recommended for the ALWR reserve offsite power source in Revision 1 to the evolutionary Requirements Document.

The staff concluded that these differences between its data and the EPRI data might be due to the fact that the NSAC-147 report was not a total listing of all partial and total losses of offsite power but rather a complete listing of all total losses of offsite power with only selected partial losses of offsite power. The staff asked EPRI to address this issue.

EPRI sent the following response in a letter dated January 10, 1992:

> The decision not to require a reserve offsite power source for the passive plants was based primarily on the very limited safety significance of loss of offsite power events for these plants. It was clear that providing an incremental improvement in the reliability of offsite power for these plants would not provide an appreciable safety or availability benefit. The relatively high conditional unavailability of such a circuit given total loss of normal offsite power was one consideration in this decision, albeit a minor one.
>
> We have not yet been able to resolve fully the apparent discrepancies between the experience base reported in NSAC-147 (and used to assess reliability parameters for offsite power) and the NRC review of experience in 1988 and 1989 (as described in the NRC Office for Analysis and Evaluation of Operational Data (AEOD) Engineering Evaluation, "Failures of Electrical Supply and Power Generation Equipment Which Disrupted Plant Function at Nuclear Power Plants"). For example, we have been unable to identify the 15 total losses of offsite power in 1988 cited in the question, nor have we been able to identify 14 such events not reported in NSAC-147. Furthermore, NSAC-147 includes losses of offsite power that are not included in the AEOD report. In any event, in the light of the comment in this question, we are reviewing the data assessment provided in Appendix A to Chapter 1 and will update Appendix A accordingly.

In the DSER for Chapter 11, the staff concluded that the 15 loss of offsite power events it referred to above were obtained from the full body of the data evaluated by AEOD in researching its report referenced by EPRI. Not all the data were used in the final AEOD report, however, so that report was not necessarily a good source for the data referenced by the staff in this issue. Plant designers should, therefore, resolve this discrepancy and should ensure that their PRA models are consistent with their design and Chapter 11 of the passive Requirements Document. The staff will pursue this issue during its review of an individual application for FDA/DC.

3.2.6 Failure Rate of Main Step-Up Transformers

In the ALWR component failure data survey in Annex A (Part 2) of Appendix A to Chapter 1, Item 75 identifies the failure rate for high-voltage transformers to be used in the passive ALWR probabilistic risk assessments as 1.2E-6 failure/hour. If this number is used for main step-up transformers, it implies an equivalent rate of slightly more than one catastrophic failure of a main transformer per year for the U.S. nuclear industry. In U.S. plants, the NRC AEOD has informally identified four main step-up transformer failures in 1989, four in 1990, and three through April 1991. As a result, the staff asked EPRI to establish a more conservative failure rate for main step-up transformers in the ALWR that was consistent with the AEOD data.

EPRI sent the following response in its January 10, 1992, letter:

> Item 75 was intended to be applied in general to transformers operating at voltages higher than 4160 volts. As the comment points

out, experience in recent years indicates that this failure rate may be too low to be applied to main step-up transformers. Consequently, a new item will be added to Annex A (for both Volumes II and III) specifically for main step-up transformers. The suggested failure rate will be 5.4E-6 per hour.

In the DSER for Chapter 11, the staff concluded that EPRI's proposed change was consistent with the staff's recommendation and was, therefore, acceptable. This was identified as a confirmatory issue.

In Revision 4, EPRI made the change; therefore, this confirmatory issue is closed.

4 MEDIUM-VOLTAGE AC DISTRIBUTION SYSTEM

4.1 Function and Description

Section 4.1.1 of Chapter 11 of the passive Requirements Document requires that the medium-voltage ac distribution system consist of the onsite electric power distribution circuits that operate at voltages ranging typically from 4.16 kV to 13.8 kV and supply power to medium-voltage loads. The system will include switchgear buses, circuit breakers, and unit substation transformers as well as their associated local instrumentation, controls, and protective relays. It also will include all buses and cables connecting the switchgear buses to their sources and loads.

The medium-voltage ac distribution system will include the top-tier (first-tier) non-safety-grade buses that will feed loads that have traditionally been non-safety related, such as reactor coolant pumps, main feedwater pumps, condensate pumps, and circulating water pumps. The source of power to these buses will be unit auxiliary transformers that will be connected to the combination of main generator, generator breaker, and main step-up transformer. This source will provide power to the loads during power operation and during shutdown conditions by opening the generator breaker and backfeeding.

The medium-voltage ac distribution system also will include the second-tier non-safety-grade buses that will feed what EPRI calls "permanent non-safety loads," which comprise loads that traditionally have been non-safety related and loads that traditionally have been safety related, such as residual heat removal (RHR) pumps, backup feedwater pumps, component cooling water pumps, and service water pumps. These loads normally will be fed from the unit auxiliary transformers, and in the event of a loss of power via these transformers, selected loads will be automatically transferred to the standby power sources. The set of loads to be automatically transferred includes loads that could facilitate operator response to the loss-of-normal-power event, and could be required to operate within 30 minutes following the onset of the event. Originally, no second offsite power circuit was specified for the second-tier buses, but, in response to a staff question, EPRI has proposed that a second offsite power circuit to these buses be required for maintenance purposes only. The staff addresses this issue in Section 3.2.1 of this chapter.

Two dedicated second-tier buses are specified for both the PWR and BWR designs to feed the permanent non-safety loads. Unless proven impractical at the time of detailed engineering, four first-tier buses will be used in PWR designs to feed the non-safety loads, and two first-tier buses will be used in BWR designs to feed the non-safety loads.

4.2 Evaluation

4.2.1 Lack of Safety-Grade Qualification for the Second-Tier Distribution System

The second-tier buses will provide power to loads that traditionally have been safety related but in the passive designs will be used primarily for backup

and pre-event support to the passive safety systems, normal system operation, and/or will provide cold-shutdown capability. Neither the loads nor the electrical distribution systems that power them, however, are qualified to safety-grade requirements. The staff addresses this issue, the RTNSS, in Section 2.2.1 of this chapter.

4.2.2 Capability of Powering the First-Tier Non-Safety Load Buses From the Second-Tier Permanent Non-Safety Load Buses

Section 4.2.6 of Chapter 11 of the passive Requirements Document originally required that the medium-voltage ac distribution system be designed to permit the energizing of most medium-voltage buses from the onsite standby power sources. Because it was unclear what was meant by "most" medium-voltage buses, the staff asked EPRI to clarify if the onsite standby power sources will be capable of supplying a portion of the first-tier non-safety loads as well as the second-tier permanent non-safety loads, or if they will be capable of only supplying both divisions of permanent non-safety loads. This requirement should more clearly define which medium-voltage buses will be required to be energized from the onsite standby power sources.

In response to the staff query, EPRI stated in a letter dated January 10, 1992:

> The purpose of the onsite standby power sources is essentially to supply power to the permanent non-safety loads following a loss of their normal power supply as indicated in Sections 5.2.1 and 5.3.1. The second-tier buses will also be provided with a separate power supply circuit for use in the event of an extended unavailability of the normal power supply, e.g., during plant outages. Section 4.2.6 of Chapter 11, Volume III will be revised as follows:
>
> Requirement:
>
> The system shall be designed to permit energizing the buses of the first tier of power distribution systems from the buses of the second tier in order to support maintenance activities in the event of a prolonged outage of the normal power supply to the first-tier buses.
>
> Rationale:
>
> The requirement for installed main step-up and auxiliary transformers will ensure that outages of the normal power supply to the first-tier buses are reduced to a minimum. Nevertheless, provisions should be made to permit energizing these buses from another source for maintenance purposes in the event of an extended outage of their normal power supply.

In the DSER for Chapter 11, the staff took note of EPRI's revised requirement for a maintenance power supply to the first-tier buses via the second-tier buses. There still remained the question, however, regarding the need for a normally available second source of power to the non-safety first-tier loads

addressed by the staff in Section 3.2.2 of this chapter. As indicated there, this aspect of a second source of power to the non-safety buses would be pursued as part of the open issue pertaining to RTNSS.

4.2.3 Use of Adjustable Speed Motor Drives

Section 4.4.5.4.4 of Chapter 2 of the passive Requirements Document requires that the feedwater pumps have adjustable speed motor drives, and Section 4.4.5.4.6 of Chapter 2 requires that the adjustable speed motors be the synchronous type with a load-commutated inverter solid-state power supply capable of providing the frequency range and power required by the pump motor in all specified modes of operation. Solid-state power supplies to heavy power loads generally create large amounts of harmonic distortion on the distribution systems that feed them. This distortion can cause problems for other equipment connected to the same distribution system. For example, Figure 11.2-2 in Chapter 11 shows that the permanent non-safety-grade main load buses (which have some safety significance) will normally be fed from the non-safety-grade load buses that also will feed the reactor feedpumps in the BWR design. The staff, therefore, asked EPRI whether consideration had been given to the effects that the distortion created by the variable speed drives can have on other equipment. EPRI sent the following response in a letter dated January 10, 1992:

> Recent plant experience with large adjustable speed motor drives indicates that proper design can suppress harmonics to acceptably low levels. Specifically, large adjustable speed drive (ASD) installations now typically include an input transformer ahead of the solid-state converter section. The input transformer can limit harmonics to under 3% total voltage distortion. IEEE Standard 519, "IEEE Guide for Harmonic Control and Reactive Compensation of Static Power Converters," establishes a limit of 5% for voltage distortion on low (480 V) and medium (2.4 kV and above) voltage power systems. Similarly, military standards limit each current harmonic to 3% of full load fundamental current in order to keep the total harmonic voltage distortion within 5% (Ref. DOD-STD-1399 and others). In order to ensure that the use of ASDs will not affect the quality of power supplied to the plant loads, a requirement will be included in Section 2.6.2 of Chapter 11 for both the evolutionary plant and the passive plant to limit the harmonic distortion caused by ASDs to a conservative value of maximum 3%. The requirement will be worded as follows:
>
> "Adjustable speed drives shall be designed to have a minimum harmonic distortion effect on the plant's electrical system voltages. Specifically, the total voltage distortion resulting from the operation of such equipment shall be limited to 3 percent."

In the DSER for Chapter 11, the staff stated that EPRI's proposed requirement to limit the total voltage distortion to 3 percent should greatly reduce the amount of equipment that is affected by the harmonic distortion. If any remaining sensitive equipment is affected by this relatively low level of

distortion, modifications to the equipment itself would likely be more appropriate than further reduction of the distribution system voltage distortion. EPRI's new requirement was, therefore, acceptable. The staff identified this as a confirmatory issue. The staff has verified that the requirement was incorporated in Revision 3; therefore, this confirmatory issue is closed.

5 ONSITE STANDBY AC POWER SUPPLY SYSTEM

5.1 Function and Description

Section 5.1 of Chapter 11 of the passive Requirements Document states that the onsite standby ac power supply system will include the onsite standby ac power sources and associated power supply circuits up to the source breakers of the onsite ac distribution systems. The standby power sources will consist of the prime movers and ac generators, the auxiliary systems (such as starting, lubrication, cooling, fuel supply, and excitation), the fuel storage and transfer system, and the associated local (close to the equipment) instrumentation and control systems. The onsite standby ac power supply system will be designed to supply ac power to the permanent non-safety loads (second-tier) in the event of a main generator trip and loss of offsite power. Section 5.1 states that operation of the onsite standby ac power supply system will not be required to ensure nuclear safety. It also states that when the plant will be in cold-shutdown conditions, the onsite standby ac power supply system can also be used to supply power during maintenance of the offsite power supply system.

Section 5.2.1 of Chapter 11 states that two onsite standby power sources will be provided to supply power to the permanent non-safety loads in the event of a turbine trip with loss of offsite power. The rationale portion of this section on the requirement for the standby power sources states it is expected that the standby power sources will consist of two identical combustion turbine or diesel engine-driven generating sets.

5.2 Evaluation

5.2.1 Capability and Capacity of the Onsite Standby Power Sources

Section 1.5.2 of Chapter 11 states that the ALWR passive plant design will provide for two relatively small onsite standby ac power sources. It originally stated that, specifically, they would be sized so that, in the event of a loss of offsite power, each would be able to supply enough power to maintain the unit in hot standby, and both, operating together, would be able to supply the power required to achieve normal cooldown and to operate most of the plant loads not strictly required for power operation. EPRI stated that each unit would have a short-term power rating greater than the sum of the loads of one division of permanent non-safety loads and safety battery chargers that could be powered concurrently during hot standby. The total short-term power generation capacity of both units operating together would be greater than the sum of the loads of both divisions of permanent non-safety loads and battery chargers that could be powered concurrently for normal plant cooldown or during plant outage.

The staff concluded that these requirements would not provide a sufficient level of safety for shutdown of the plant and contradicted statements made in Section 2.5.6.2.2 of Appendix B to Chapter 1.

Because the non-safety ac systems would be the normal means used to bring the plant to cold shutdown conditions, the staff concluded that the ac power systems should be designed as indicated in Section 2.5.6.2.2 of Appendix B to

Chapter 1; that is, there should be no single failure of these systems "which would result in the inability to terminate use of the passive safety-grade system and achieve cold shutdown, if desired." Each of the standby power sources should, therefore, have the capability and capacity to power the necessary complement of equipment to bring the plant to a cold-shutdown condition and maintain it there. Because the typical short-term power ratings of generation equipment may not be sufficiently long to power the equipment needed to bring the plant to a cold-shutdown condition and to allow for some load growth, the continuous power rating of the standby power source should be used in this sizing criterion. The staff related this position to EPRI, and EPRI responded in a letter dated January 10, 1992. The staff provided the details of EPRI's response in the DSER.

In the DSER for Chapter 11, the staff stated that EPRI's response, which included proposed revisions to the passive Requirement Document, provided necessary clarification of EPRI's requirements in these areas. It was clear from the proposed changes that the passive ALWRs will have the capability to bring the plant to a cold-shutdown condition utilizing the non-safety ac electrical power systems assuming the most limiting single active failure in the non-safety ac electrical power systems or of the onsite standby power sources coincident with a loss of normal ac (offsite) power. On the basis of EPRI's proposed requirements, however, these systems would not necessarily be required to have this capability assuming failures, such as a cable fault, because the EPRI requirements did not define cable faults as active failures. The staff stated that it would address this portion of the issue following final resolution of the RTNSS issue discussed in Section 2.2.1 of this chapter. The staff also stated that although the active non-safety ac systems were only specified to have the capability to handle a pipe break area equivalent to that of a 3/8-inch (inside diameter) pipe, EPRI should clarify whether the non-safety ac electrical power systems would be required to support this capability assuming a single failure. This was identified as an open issue.

In its response dated September 29, 1992, EPRI submitted a clarifying change to Section 5.3.1.1 of Chapter 11 that reads as follows:

> Consistent with Chapter 3, Section 2.3.2.2, the non-safety electrical power systems will be required to support the active non-safety systems so they will maintain the capability to handle a pipe break area equivalent to that of a 3/8-inch pipe, assuming a single active failure or loss of design function of a single passive electric component in the non-safety electric power system. The plant designer shall verify that the standby power source rating is adequate to accommodate this operational mode while meeting the above sizing requirements.

The above change to the passive Requirements Document, as well as the change identified by the staff in Section 2.2.1 of this chapter, indicates that EPRI's intent is to support the redundant non-safety auxiliary systems identified in Section 2.3.3.2 of Chapter 3 (which includes the systems necessary to handle the 3/8-inch pipe break) with redundant electrical systems that will maintain their support capability given a loss of function of <u>any</u> one of their components. These changes have all been incorporated by Revision 4 into the passive Requirements Document. The staff concludes that these

changes satisfactorily clarify EPRI's requirements on this issue; therefore, the portion relating to the staff's discussion of cable faults and passive failures in Chapter 11, Section 5.2.1, of the DSER is closed.

Also, it was not clear to the staff from EPRI's proposed changes why the requirement identified in the first bullet of Section 5.3.1.1 did not totally envelope the requirement identified in the second bullet. The changes made to Section 1.5.2 in Chapter 11 seem to imply that these are two separate and distinct capabilities. The staff stated in the DSER that EPRI should clarify this point and this identified this an open issue.

In its letter dated September 29, 1992, EPRI responded that the first bullet requires that one standby power source would be enough to bring the plant to cold shutdown and maintain it in that situation; while the second bullet requires that with the two standby power sources it would be possible to bring the plant to refueling condition, while supplying power to most of the loads not strictly required for power generation. EPRI further stated that Section 5.3.1.1 specifies that the standby power sources will be designed to meet the most limiting of these requirements and this is consistent with Chapter 11, Section 1.5.2.

The staff agrees that Section 5.3.1.1, specifies that the standby power sources must meet the most limiting of the subject criteria. Therefore, the question of whether one criterion envelopes the other is not a fundamental concern and the staff will not pursue it further; this open issue is closed.

In addition, with regard to EPRI's proposed revision to Section 5.3.1.1, the staff addressed a separate question to EPRI recommending that at least a 20-percent margin be specified for the continuous power rating of the standby power sources to allow for the addition of unanticipated loads. In its January 10, 1992, letter, EPRI responded that the last sentence of the requirement in Section 5.3.1.1 would be revised as follows: "The short-term power rating shall include sufficient margin to accommodate some possible load growth. This margin shall be in the range of 10 to 15 percent."

The staff stated, in the DSER, that it was not clear how this last revision would tie in with the previously identified revision to Section 5.3.1.1. It appeared that EPRI was proposing to use a short-term power rating of 2000 hours (with margin) to determine the size of the diesel generators rather than the continuous power rating (with margin) as recommended by the staff. The current version (1984) of IEEE 387, "IEEE Standard Criteria for Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations," however, only defines a short-time rating for diesel generators of 2 hours. This rating is defined as the electric power output capability that the diesel generator unit can maintain in the service environment for 2 hours in any 24-hour period without exceeding the manufacturer's design limits and without reducing the maintenance interval established for the continuous rating. The 2000-hour rating that was used in the past, on the other hand, defined the power rating and length of time operation might be accumulated at that power level before substantial inspection and maintenance had to be performed. The staff was concerned that if the standby units are used for peaking generation (see Section 5.2.3 below), a substantial portion of the 2000 hours could be used up during peaking operation leaving little time available for standby operation. In the DSER for Chapter 11, the staff stated

that EPRI should clarify whether the 2000-hour rating was still applicable to current diesel generators and combustion turbine-generators, describe how peaking operation might affect the 2000-hour rating and standby operation, and clarify how the margin specified is to be applied to the machine rating. This was identified as an open issue.

In its response dated September 29, 1992, EPRI stated that this issue was related to a Chapter 11 issue in Section 5.2.3 below, which in turn was related to the RTNSS issue. EPRI indicated that the industry position would be established according to the resolution of that issue.

The staff agrees that this issue is a good candidate to be addressed under the RTNSS issue it discusses in Section 2.2.1 of this chapter. To resolve this issue, the staff expects the standby power sources to be addressed during the design-specific RTNSS process described in SECY-94-084. If the standby power sources are determined to be risk significant under this process, the staff will pursue this issue further. It will, therefore, address this issue during its review of an individual application for FDA/DC. Therefore, this open issue is closed.

5.2.2 Location of the Non-Safety Standby AC Power Sources

Section 2.3.3.6 of Chapter 1 and the rationale portion of Section 2.3.4 of Chapter 11 allow one of the non-safety standby ac power sources to be located in the vicinity of the plant rather than on the plant site. Because of the reliance on the non-safety ac electrical power system to bring the plant to a cold-shutdown condition and provide backup to the passive safety systems and because of the vulnerability of power sources located off site, the staff indicated to EPRI that both non-safety ac standby power sources (whether combustion turbine-generators or diesel generators) should be located on site. In a letter dated January 10, 1992, EPRI committed to revise Chapters 1 and 11 to require that both non-safety standby ac power sources be located on site without exception. The staff identified this as a confirmatory issue in the DSER for Chapter 11. These changes were incorporated into the passive Requirements Document Revisions 3 and 4; therefore, this confirmatory issue is closed.

5.2.3 Peaking Operation of the Non-Safety Standby Power Sources

The requirements and rationale portions of Sections 4.2.5, 5.3.1.1, and 5.8.1 of Chapter 11 and Section 4.62 of Appendix B to Chapter 1 address peaking operation of the non-safety standby ac power sources. EPRI anticipates that some plant owners will utilize the standby power sources for peaking generation and that some of the specified requirements will facilitate such use. The use of the standby power sources for peaking generation, however, could make them vulnerable to common-cause problems with the offsite power system such as grid perturbations and system faults. The result would be a potentially higher frequency of station blackout (SBO). The staff, therefore, indicated to EPRI that peaking operation of the non-safety standby ac power sources should not be allowed.

In a letter dated January 10, 1992, EPRI responded that SBO was not an issue for the ALWR passive plant, and adequate protective devices and load rejection capability were required to separate the onsite units from the offsite system.

EPRI indicated that peaking operation was considered acceptable for the onsite standby units of the ALWR passive plant as much as it was for the alternate ac power sources of the evolutionary plant and existing plants.

In the DSER for Chapter 11, the staff agreed that peaking operation of the non-safety standby ac power sources was acceptable in principle for the passive ALWRs because they generally play a less important role than do the standby power sources in existing plant designs. However, the staff indicated that certain provisions should first be made in the design for peaking operation, and limitations should be placed on its use. Peaking operation should only be allowed for standby units that have substantially more capability than that needed to power their associated division of permanent non-safety loads, and only one standby unit at a time should be used for peaking. The additional capacity would make the standby units more useful as peaking units, would help ensure accumulated hours of peaking operation use do not severely reduce hours available for emergency operation (see Section 5.2.1 of this chapter), and would enhance their capability as an emergency power source during non-peaking operation by providing capability to power more than one division of permanent non-safety loads if needed. Operating only one unit at a time as a peaking unit ensures that one unit will remain free from offsite system failures. Sufficient protective features should be installed to protect the standby units from offsite system perturbations or failures when paralleled with the grid, and provisions should be made to accommodate the higher available fault currents from the larger standby units when connected to the onsite systems. The one-line distribution system diagrams shown in Chapter 11, Figures 11.2-1 and 11.2-2, may also not be appropriate for a system that will utilize the standby power sources as peaking units. This was identified as an open issue. Also, the acceptance in principle of the standby power sources for peaking generation at passive ALWRs was made with the provision that no overriding staff concern would be identified in the RTNSS issue (Section 2.2.1 of this chapter) that would make their use in this manner inappropriate.

In its letter dated September 29, 1992, EPRI responded that since the peaking operation mode issue was related to the RTNSS issue, the industry position would be established according to the resolution of that generic issue.

The staff agrees that this issue is a good candidate to be addressed under the RTNSS issue it discusses in Section 2.2.1 of this chapter. To resolve this issue, the staff expects the standby power sources to be addressed during the design-specific RTNSS process described in SECY-94-084. If the standby power sources are determined to be risk significant under this process, the staff will pursue this issue further. It will address this issue during its review of an individual application for FDA/DC. Therefore, this open issue is closed.

5.2.4 Reliability of the Non-Safety AC Standby Power Sources

Section 5.2.2.1 of Chapter 11 states that, for the purpose of evaluating the reliability of the ac power supply system of the ALWR passive plant, the plant design should assume that the overall reliability of one standby power source is 0.9. In the rationale portion of this section, EPRI states that this specified overall reliability of standby sources includes startup and load-run reliability as well as maintenance unavailability. The staff asked EPRI how

the 0.9 overall reliability compared to the overall reliability of a typical safety-grade standby power source used in current operating plants. The staff also asked EPRI to provide the unavailability of the standby power system in the passive ALWR when the overall reliability of each standby power source is 0.9, and asked how this compared to the unavailability of standby power in current operating plants. EPRI sent the following response in a January 10, 1992, letter:

> In developing the passive Requirements Document for the passive plant, the requirements for the standby sources were not based on the safety applications of emergency power sources at existing plants. Since there is no need for safety grade ac power beyond that supplied through the dc system, the passive plant standby sources have a different role than that of the emergency power sources in current operating plants. This is discussed in more detail in the response to Question 435.1. Thus the performance requirements of the passive plant standby sources were defined considering industrial standby power sources used for investment protection and plant availability purposes and alternate ac sources (AACs) of existing nuclear plants.
>
> For the purpose of reliability evaluations, the assumed 0.9 total availability for a standby source includes unavailability due to maintenance which is assumed larger (factor 2-3) than that indicated for existing nuclear plant safety diesels. This additional unavailability allowance is provided to improve the overall quality and efficiency of maintenance activities by eliminating unnecessarily tight schedule constraints. In addition, the assumed probability of failure to start on demand for the passive plant standby sources is also assumed greater than that assumed by safety sources. The assumed failure to start probability was selected to reduce the intensity of testing programs that are required to demonstrate higher starting reliabilities.
>
> With regard to forecasting expected performance of the standby sources, it is noted that they will meet the design requirements of Section 5, including, for example, Section 5.2.2. In addition, because of an increase in the allowed starting time, reduced testing frequency, Reliability Assurance Program and more time for maintenance, it is anticipated that the availability assumptions of Section 5.2.2.1 in Chapter 11, Volume III will prove to be particularly conservative.

With regard to EPRI's opening statement that the requirements for the standby sources for the passive plant were not based on the safety applications of emergency power sources at existing plants, the staff noted in the DSER for Chapter 11 that, while Chapter 11 did not specify standby source reliability criteria equivalent to those of existing plants, Appendix A to Chapter 1 did recommend start and run failure rates for diesel generators that were identical to the rates for the evolutionary plant and were based on existing plant data. The staff also noted that the failure data in Appendix A to Chapter 1 for the combustion turbine-generator were not consistent with the Chapter 11 standby power source reliability criteria. In the rationale portion of Section 5.2.2.1 in Chapter 11, for example, EPRI stated that a maintenance outage

of 2 weeks per year, 1.2E-2/hour failure to run, and 0.95/demand start-reliability were consistent with a 0.9 overall reliability. The respective failure data for a combustion turbine-generator in Appendix A to Chapter 1, however, was a maintenance outage of 11 days per year, 1.0E-5/hour failure to run after starting, and 0.975/demand start-reliability. Use of these data in the design-specific passive ALWR probabilistic risk assessments would result in an artificially high level of reliability for the passive plant onsite ac power system that was inconsistent with the Chapter 11 standby power source reliability criteria. The staff concluded in its DSER that EPRI should use failure and unavailability rates identified in Appendix A to Chapter 1 for diesel and combustion turbine-generators that were consistent with the Chapter 11 standby power source reliability criteria. This was identified as an open issue. In addition, with regard to the acceptability of the 0.9 value for the overall reliability of each standby power source, the staff stated that it would address this issue after the RTNSS issue discussed in Section 2.2.1 of this chapter was resolved.

In its letter dated September 29, 1992, EPRI responded that the data in the different sections of the passive Requirements Document would be made consistent. Subsequently, in Revision 4 to the passive Requirements Document, some of the data in Annex A of Appendix A to Chapter 1 were revised. The data in Annex A for the standby sources are generally consistent with a 0.9 overall reliability depending on assumed surveillance intervals and mission times. However, the broader question of the reliability required of the standby sources in a passive design, and how this will be ensured (regulatory control or otherwise), need to be addressed.

This issue is a good candidate to be addressed under the RTNSS issue the staff discusses in Section 2.2.1 of this chapter. To resolve of this issue, the staff expects the standby power sources to be addressed during the design-specific RTNSS process described in SECY-94-084. If standby power sources are determined to be risk significant under this process, the staff will pursue further the issue of the required reliability of the standby power sources. It will address this issue during its review of an individual application for FDA/DC. Therefore, this open issue is closed.

5.2.5 Start Times of the Non-Safety AC Standby Power Sources

Section 5.3.2.1 of Chapter 11 requires that each standby power source automatically start, accelerate to rated speed, reach nominal voltage, and be ready to accept load within 2 minutes of receipt of a start signal. The staff asked EPRI whether 2 minutes was fast enough to preclude actuation of the passive systems following system transients or small loss-of-coolant accidents (LOCAs), particularly when the standby power system will be the only automatically available ac power system following loss of normal ac power. The staff determined that it would be better to utilize the non-safety ac systems to deal with transients or small LOCAs to the extent practical to avoid challenging the passive systems, especially if it would result in the actuation of a one-time device such as a squib-actuated valve. EPRI gave the following response in a January 10, 1992, letter:

> The non-safety inventory control systems (the CRD [control rod drive] pumps for the BWR and the CVCS [chemical and volume control

system] pumps for the PWR) are designed to handle a loss of inventory up to the equivalent of a 3/8-inch pipe break (see Chapter 3, Section 2.3.2.2). The two minute start time is sufficient to assure the non-safety inventory control system would be functioning before reaching the actuation point of passive safety injection for this rate of inventory loss. Therefore, a quicker start time for the standby power source would not reduce the number of scenarios which would actuate passive safety injection.

For the BWR, one of the CRD pumps is running (for control rod drive functions) during normal plant operation and the other pump is started automatically on a reactor vessel water level signal. Based on a 3/8-inch pipe break, it would take approximately 50 minutes for the inventory to fall (assuming loss of normal feedwater at power and loss of power to the CRD pumps) from the normal level to the actuation level of the passive inventory control system. Non LOCA-transient events which require decay heat removal are handled by the isolation condensers which provide cooling while maintaining inventory.

For the PWR, the CVCS pumps are not normally running but one is started automatically on a pressurizer water level signal. Passive safety injection does not require use of a one-time device and will not result in automatic depressurization for an extended period of time (about one hour) assuming a 3/8-inch break rate of inventory loss. A pressurizer water level signal will cause the valves on the lines from the core makeup tank (CMT) to open to provide some makeup until (after about one hour) water level has dropped to the CMT actuation level of the depressurization system.

In summary, for both designs there is no significant advantage to a quicker start time for the onsite ac power units.

The staff also asked EPRI about the lack of a requirement for automatic startup of the non-safety ac standby power sources on a LOCA signal only. This feature, which is incorporated into the design of existing plants, automatically starts the diesels on a LOCA signal and keeps them operating in standby so that they would be immediately available if a subsequent loss of offsite power (LOOP) were to occur. In the rationale portion of Section 5.3.3.2 of Chapter 11, EPRI states that automatic startup on LOCA signal only, that is, without LOOP, is not considered justified. Although the staff recognized that the non-safety ac systems are only specified to have the capability to handle 3/8-inch pipe breaks, it felt that the long startup time (2 min.) allowed for the standby power sources might increase reliance on the passive systems given a small LOCA with a delayed LOOP. The staff, therefore, asked EPRI whether the automatic start feature on a LOCA signal might be desirable in order to avoid the actuation of one-time devices, such as squib-actuated valves, in the passive systems. EPRI gave the following response in a January 10, 1992, letter:

The ALWR requirements for avoidance of actuation of the depressurization system are stated in Chapter 5, Section 1.2.1.2. It is not

expected that automatic start of the onsite power units would have a significant effect on the probability of depressurization events and therefore automatic start is not justified.

The LOCA level signal is considerably below the normal operating level. It takes from about 50 minutes to 1 hour for water level to fall to this point, assuming a small (i.e., 3/8") LOCA has occurred. These are the levels at which the reactor is depressurized and for the BWR the squib valves are opened. The non-safety inventory control pumps (e.g., CVCS and CRD) which could be supplied from the onsite standby power units in the case of a LOOP, are too small to be of much value at this point. These pumps normally control coolant inventory at a much higher level. In the case of a LOOP they will be restarted automatically to maintain level within limits as soon as power is restored. On this basis, an automatic start of the on-site standby power units upon receipt of a LOCA signal is not warranted.

On the basis of EPRI's response to the above two items, the staff agrees that there appears to be little advantage to requiring a standby power source start time of less than 2 minutes or requiring an automatic start of the standby power sources on a LOCA signal without a LOOP. This conclusion was made in the DSER for Chapter 11 with the provision that no overriding staff concern was identified in the RTNSS issue (Section 2.2.1 of this chapter) that might warrant a shorter starting time or automatic LOCA start of the standby power sources.

5.2.6 Standby Power Source Auxiliary Support Systems

Each standby power source will have the following auxiliary systems:

- starting system
- combustion air intake and exhaust system
- cooling system
- lubrication system
- fuel oil storage and transfer system

A discussion of the design criteria proposed by EPRI for these systems follows. The staff evaluated these systems against the guidelines in the following sections of the Standard Review Plan (SRP, NUREG-0800): 9.5.4, "Emergency Diesel Engine Fuel Oil Storage"; 9.5.5, "Emergency Diesel Engine Cooling Water System"; 9.5.6, "Emergency Diesel Engine Starting System"; 9.5.7, "Emergency Diesel Engine Lubrication System"; and 9.5.8, "Emergency Diesel Engine Combustion Air Intake and Exhaust System."

Starting System

The design function of the standby power source starting system is to provide a reliable method for starting the standby power sources for all modes of operation.

In Section 5.5.2 of Chapter 11, EPRI establishes the following key requirements for the starting system:

- Each standby power source will be provided with at least one compressed-air or battery-powered starting system.

- Each air starting system will be sized for three consecutive starts without recharging.

- To avoid corrosion or scaling problems, each air starting system will be provided with air dryers and air filters, and the piping material will be stainless steel or copper.

The staff concludes that the requirements established in the passive Requirements Document for the standby power source starting system, although designated as non-safety grade, are similar to the current regulatory requirements for emergency diesel generators. However, in the DSER, the staff identified the design details and classification of the standby power source starting systems as a vendor-specific item.

In its letter dated September 29, 1992, EPRI agreed that the NRC should review the diesel generators at the design-specific stage. It stated that the current regulatory requirements for Class 1E diesel generators do not apply to the onsite standby power sources because they are non-safety-related systems. On the basis of its review of EPRI's position, the staff has determined that the regulatory treatment of non-safety standby power source support systems will be in accordance with the resolution of Policy Issue III.A, Regulatory Treatment of "Non-Safety Systems for Passive ALWRs," which was identified in Annex A of Appendix B to Chapter 1 of the DSER. In SECY-94-084, the staff agrees with EPRI on a process for resolving this policy issue. In the process, the ALWR designer will identify all risk significant non-safety SSCs functions and their corresponding reliability/availability (R/A) missions. Furthermore, the designer will propose regulatory treatment appropriate to provide confidence that the identified R/A mission will be achieved. The staff will evaluate the results of this process during its review of an individual application for FDA/DC to determine if enhanced regulatory oversight is required for these systems.

Combustion Air Intake and Exhaust System

The basic function of the standby power source combustion air intake and exhaust system is to supply combustion air of suitable quality to the standby power source and to exhaust the combustion products from the standby power source to the atmosphere.

In Section 5.5.3 of Chapter 11, EPRI establishes the following key requirements for the combustion air intake and exhaust system:

- Each standby power source will be provided with an independent combustion air intake and exhaust system. The system will be sized and physically arranged so that no degradation of engine function will be experienced when the unit is required to operate continuously at its maximum rated power output.

- Each combustion air intake system will be provided with means of reducing airborne particulate material entering the system, assuming the maximum expected airborne particulate concentration at the combustion air intake.

- The arrangement and location of the combustion air intake and exhaust structures will be such as to preclude a reduction of engine power output due to intake of exhaust gases or other diluents (e.g., fire suppression agents) that could reduce oxygen content below acceptable levels.

- The components of the combustion air intake and exhaust system that are exposed to atmospheric conditions will be protected from possible clogging due to such conditions as ice, snow, and dust.

The staff concludes that the requirements established in the passive Requirements Document for the standby power source combustion air intake and exhaust system do not conflict with regulatory requirements and are, therefore, acceptable.

Cooling System

The design function of the standby power source cooling system is to maintain the temperature of its associated diesel engine within a safe operating range under all load conditions and to keep the engine lubricant preheated during standby conditions to improve starting reliability.

In Sections 5.5.1 and 5.5.5 of Chapter 11, EPRI establishes the following key requirements for the cooling system:

- Each standby power source will have its own independent cooling system, which will not rely on the operation of the plant's component cooling or service water systems.

- Each standby power source will be provided with a prewarming system and a pre-/post-lubrication system, as necessary, to maintain the unit in a standby mode and ensure the starting capability specified in Section 5.3.2 of Chapter 11.

In the DSER for Chapter 11, the staff noted that for the evolutionary plant, EPRI requires a water-to-water cooling system for the standby power source units because of its simplicity and reliability. A totally independent air cooling system (with radiators and engine-driven fans) was considered for the evolutionary plant and was found to be (1) less reliable as a result of recurrent vibration problems caused by the fans, (2) more complex because of the large air flows required to cool large standby power source units (large air flows result in large radiators with a large number of fans, and large intake and ducting structures that have to be adequately protected), and (3) susceptible to atmospheric and climatic conditions (such as large temperature variations and tornadoes). A passive plant combustion turbine will be provided with an air cooling system with radiator and ac motor-driven fans for oil cooling because of the simpler cooling requirements. Therefore, in the DSER, the staff stated that given that the standby ac power source for the EPRI passive design would be approximately the same size as current standby power sources, EPRI should justify the use of independent self-contained cooling systems for the standby power source. The staff identified this as an open issue.

In its letter dated September 29, 1992, EPRI stated that for the smaller size units to be used in the passive plants (4000 MWe versus 5500 MWe), both

manufacturers and operating experience report that the self-contained cooling systems are equally good. Furthermore, the self-contained systems provide independence from external systems. On the basis of this response, the staff's concern regarding the capability of an independent self-contained cooling system to provide adequate cooling to the standby power source is resolved. Therefore, this open issue is closed.

Lubrication System

The basic function of the standby power source lubrication system, which is an integral part of the engine, is to provide essential lubrication and cooling for the components of the engines.

In Section 5.5.6 of Chapter 11, EPRI establishes the following key requirements for the lubrication system:

- Each standby power source will have its own independent lubrication system and will be equipped with a set of engine-driven pumps.

- Each standby power source will be provided with a pre-/post-lubrication system consisting of an ac motor-driven pump and a backup dc motor-driven pump designed to ensure continuous prelubrication while the standby power source is in the standby mode. Transfer from the normal ac motor-driven pump to the backup dc motor-driven pump will be automatic in case of pump failure or loss of ac power.

The staff concludes that the requirements established in the passive Requirements Document for the standby power source lubrication system do not conflict with regulatory requirements and are, therefore, acceptable.

Fuel Oil Storage and Transfer System

The basic function of the standby power source fuel oil storage and transfer system is to provide a separate and independent fuel oil supply train for each standby power source and to permit operation of the standby power source at full load for a minimum of 7 days without replenishing fuel.

In Section 5.5.10 of Chapter 11, EPRI establishes the following key requirements for the standby power source fuel system:

- Each standby power source will be provided with an independent fuel supply system in order to prevent a single failure from affecting more than one unit.

- Each fuel supply system will be provided with fuel filters and water separators in the supply lines to ensure fuel quality.

- Each fuel oil storage tank will be sized to support operation of the associated standby power source at its maximum continuous rating for a minimum of 7 days.

- Each day tank will have enough capability to operate its associated standby power source for at least 4 hours at its maximum rated capacity.

- Each fuel oil tank will have the capacity to be tested for the presence of water and, if necessary, to be drained of water from the tank bottom.

- Adequate access will be provided for sampling fuel oil throughout the fuel supply system.

The passive Requirements Document does not define the detailed design features and administrative controls required to maintain the quality of the stored fuel oil, to protect the supply lines from contamination, or to minimize fire hazard during and after filling operations. These detailed design features and administrative procedures are expected to be defined by the plant designer in consultation with the plant owner and operator on the basis of the particular arrangement of the fuel oil storage and transfer system. In addition, the design of the filters and the location of the transfer pumps have not been defined. The final system design will ensure that the pumps will have sufficient head and capacity to transfer fuel oil to the day tanks as required.

The staff concludes that the design requirements established in the passive Requirements Document for the standby power source fuel oil storage and transfer system do not conflict with regulatory requirements and are, therefore, acceptable. However, in the DSER, the staff identified the design details and the classification of the standby power source fuel oil storage and transfer system as a vendor-specific item.

In its letter dated September 29, 1992, EPRI agreed that the NRC should review the diesel generator at the design-specific stage. It stated that the current regulatory requirements for Class 1E diesel generators do not apply to the on-site standby power sources because they are non-safety-related systems. On the basis of its review of EPRI's position, the staff has determined that the regulatory treatment of non-safety standby power source support systems will be in accordance with the resolution of Policy Issue III.A, RTNSS which was identified in Annex A of Appendix B to Chapter 1 of the DSER. In SECY-94-084, the staff agrees with EPRI on a process for resolving policy issue. In the process, ALWR designers will identify all risk-significant non-safety SSCs functions and their corresponding R/A missions. Furthermore, the designer will propose regulatory treatment appropriate to provide confidence that the identified R/A mission will be achieved. The staff will evaluate the results of this process during its review of an individual application for FDA/DC to determine if enhanced regulatory oversight is required for these systems.

6 LOW-VOLTAGE AC DISTRIBUTION SYSTEM

6.1 Function and Description

Section 6.1 of Chapter 11 of the passive Requirements Document states that the low-voltage ac distribution system will consist of the onsite electric power distribution circuits that supply power to plant loads at 600 V or less. The system will not include the low-voltage vital ac power supply system or the normal and emergency lighting systems. These are covered in Sections 7 and 8 of Chapter 11 of the passive Requirements Document. The system will include load centers, motor control centers (MCCs), distribution transformers, and distribution panels, as well as the associated protective relaying and local instrumentation and controls. Also included will be the cables, connections, and electrical penetrations used throughout the system. A limited portion of the system will be safety related.

The low-voltage ac distribution system will be fed from the medium-voltage ac distribution system's first-tier non-safety load buses and the second-tier permanent non-safety load buses described in Section 4.1 of this chapter. Like the medium-voltage ac buses that will supply them, the low-voltage ac distribution buses will power loads that have traditionally been non-safety related and loads that have traditionally been safety related. In addition, the low-voltage ac distribution system will feed the Class 1E battery chargers that normally power the safety related portions of the dc and low-voltage vital ac power supply systems. The system is required to be designed to provide the capability of establishing simple connections to supply power to the safety divisions of the dc and low-voltage vital ac power supply systems from transportable ac power generators. The switchgear to make this connection is required to be qualified as Class 1E and to meet the same physical separation and electrical independence requirements as the safety divisions.

Section 6.2.2 of Chapter 11 specifies that the low-voltage ac power distribution system will be designed so that failure or unavailability of a single unit substation or distribution transformer will not preclude continuous system operation. Section 6.3.2 states that, for the non-safety portions of the low-voltage ac distribution system, redundant loads will generally be fed from separate load centers or MCCs, and load centers will generally be double ended with provisions to receive power from both feeders. Cross-ties will be allowed on the non-safety portions of the system.

6.2 Evaluation

Thermal Overload Protection of Valve Motor Operators

Section 6.5.2 of Chapter 11 of the passive Requirements Document addresses thermal overload devices used to protect valve motor operators. However, the requirement for the passive ALWR does not address with Class 1E valve motor operators as is done for the evolutionary ALWR. The evolutionary ALWR requirements state that the Class 1E thermal overload devices will be set so as not to compromise the safety function of the system. Although the staff recognized that EPRI did not include requirements for Class 1E valve motor operators in Section 6 of Chapter 11 because most of the low-voltage ac distribution system is not qualified as Class 1E, it recommended to EPRI

that the more conservative requirements be specified because of the support role some of the non-safety ac systems play to provide backup to the passive safety systems and bring the plant to cold shutdown.

In its January 10, 1992, letter, EPRI responded that although the non-safety ac systems are not needed to support the passive safety systems following an event, it is considered appropriate to require that the thermal overload devices be carefully sized.

Therefore, EPRI committed to add the following sentence to Section 6.5.2 of Chapter 11: "The thermal overload devices shall be carefully selected and sized so as to provide the necessary motor-operator protection while minimizing the probability of spurious interruptions of valve actuation."

In the DSER for Chapter 11, the staff concluded that this change satisfied the intent of its recommendation and identified this as a confirmatory issue.

EPRI has added the sentence to Revision 3; therefore, this issue is closed.

7 DC AND LOW-VOLTAGE VITAL AC POWER SUPPLY SYSTEMS

7.1 Functions and Key Design Requirements

Section 7.1 of Chapter 11 of the passive Requirements Document states that the dc power supply system will consist of the electric power supply and distribution equipment and circuits that will provide dc power to the plant dc loads. The system will begin at the source terminals of the safety and non-safety battery chargers. It will end at the input terminals of the inverters of the low-voltage vital ac power supply system. The low-voltage vital ac power supply system will consist of the electric power supply and distribution equipment and circuits that will provide low-voltage ac power for continuous operation of safety instrument loads, computer systems, and other important plant loads. The system will begin at the input terminals of the inverters and backup regulating transformers and will end at the input terminals of the system's load.

The system will include battery sets, battery chargers, inverters, regulating step-down transformers, motor control centers, distribution panels, associated protective relays and instrumentation, and all cabling and wiring from the source terminals up to the terminals of the system loads.

The dc power supply systems and associated vital ac power systems in the passive ALWRs will include both safety-related and non-safety-related power supplies that will provide support to the types of loads traditionally powered by these systems in existing active plant designs; in addition, they will provide power to support the operation of the passive safety systems for 72 hours. The safety-related portion of the dc power supply system will provide the power to support the operation of the passive safety systems for 72 hours, as well as power to the safety-related vital low-voltage ac inverters that will power the reactor protection and safety systems actuation channels. The following major provisions are specified in Section 7 of Chapter 11 of the passive Requirements Document for the dc and low-voltage ac power supply systems:

- The dc power supply system is required to be designed with sufficient redundancy to ensure that, in the case of loss of offsite power concurrent with a turbine trip, the loss of any plant battery or dc bus concurrent with a single independent failure in any other system required for shutdown cooling will not result in a total loss of reactor cooling capability.

- The dc and low-voltage vital ac power supply systems are required to be designed with sufficient redundancy to ensure the following:

  - The failure or unavailability of a single battery, battery charger, or inverter will not result in a plant trip or a forced outage.

  - Battery equalization may be performed off-line and each battery, battery charger, or inverter may be separately tested and maintained (including battery discharge tests, battery cell replacement, battery charger and inverter replacement), without affecting continuous plant operation at 100-percent power.

- The batteries of the dc power supply system are required to be sized to meet the following operational requirements:

  - to supply power to their loads for a period of at least 2 hours based on the most limiting load profile without load shedding

  - to permit operating selected safety loads for at least 72 hours following a licensing-basis event (which includes the loss of all ac power sources) without manual load shedding or load management program

- Each dc bus is required to be normally connected to a battery and a battery charger. In addition, provision shall be made to permit connecting each bus to a standby, backup dc source (i.e., a combination of a battery and a battery charger). This backup source is required to have sufficient capacity to permit continuous plant operation at 100-percent power in case of failure or unavailability of the normal battery or battery charger.

- Uninterruptible power supplies are required to be provided for operation of low-voltage vital ac safety loads, including reactor protection and safety systems actuation channels, and safety systems instrumentation and control loads.

- Each uninterruptible power supply normally is required to be fed from a dc bus so as to eliminate the need for dedicated ac sources and rectifiers.

- Each uninterruptible power supply is required to be provided with a backup ac source, consisting typically of a regulating transformer, with sufficient capacity to allow normal system operation in case of failure or unavailability of a single inverter. For the Class 1E uninterruptible power supplies, the backup sources are required to be qualified as Class 1E.

- Each uninterruptible power supply is required to be provided with make-before-break static switches for automatic transfer of the loads to the backup source of ac power on failure of the inverter. Manual switches must be provided for manual transfer to the backup source for maintenance of the inverter or the static switch.

- Each reactor protection and safety systems actuation channel is required to be powered from a dedicated Class 1E dc source that is normally independent of the dc sources of the other channels and of other dc sources. Each source will include a battery and battery charger capable of supplying power to its associated reactor protection system channel for a minimum of 2 hours.

- The non-safety portions of the dc and low-voltage vital ac power supply systems are considered part of the permanent non-safety systems. As such, they must be energized from an ac source as long as power from the normal power source or an onsite standby power source is available.

- A separate 250-V dc non-safety dc power supply system is required to be provided for operation of the plant's large dc loads such as standby lube oil pumps or seal oil pumps.

- The non-safety 125-V dc power supply will include redundant power supply buses, batteries, and battery chargers. Redundant chargers are required to be fed from separate ac buses.

- Uninterruptible power supplies are required to be provided for operation of non-safety low-voltage vital ac systems, including plant computers, instrumentation and control loads, and fire-detection systems.

- The security and plant lighting systems also will receive normal power from the dc and low-voltage ac power supply.

7.2 Evaluation

7.2.1 Loss of Power to a DC Bus

Section 7.2.2 of Chapter 11 required that the dc and low-voltage vital ac power supply systems be designed with sufficient redundancy to ensure that the failure or unavailability of a single battery, battery charger, or inverter would not result in a plant trip or a forced outage. In its DSER for Chapter 11, the staff stated this requirement should also apply to the loss of a dc bus. That is, the loss of a dc bus should not result in a plant trip or a need for a manual trip to stabilize an abnormal or unstable plant condition. The staff did not interpret this requirement to include shutdowns (or "forced outages") that may result from complying with a technical specifications "action" statement. The staff's concern was that loss of power to a dc bus could cause a transient or plant trip requiring the response of safety systems, but those responses could fail because of a subsequent loss of power to an additional dc or ac bus or loss of an additional dc- or ac-powered load.

The staff had also identified this issue in its review of the evolutionary Requirements Document and concluded that some potential existed that not all losses of dc buses had been addressed. In its DSER for Chapter 11 of the evolutionary Requirements Document, the staff concluded, therefore, that the requirements of the evolutionary Requirements Document were insufficient to ensure that loss of an electrical bus would not result in a plant transient and simultaneous loss of single-failure protection in any safety-related system. Because the requirements in the passive Requirements Document were essentially identical to those for the evolutionary plant in this area, the staff drew the same conclusion for it as well. The staff also noted that Table B.1-2 in Appendix B to Chapter 1 of the passive Requirements Document commits to meet RG 1.153, "Criteria for Power, Instrumentation, and Control Portions of Safety Systems," which endorses IEEE 603-1980, "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations," as providing methods acceptable to the staff for complying with the Commission's regulations with regard to the design of the power portions of the safety systems. IEEE 603-1980 provides single-failure criteria in Section 5.1 that reflect the staff's position in this matter. The staff stated in the DSER that the passive Requirements Document should, therefore, contain sufficient requirements to ensure that loss of an electrical bus (either ac or dc) would

not result in a plant transient and simultaneous loss of single-failure protection in any safety-related system needed to respond to the event. This was identified as an open issue.

In its response dated September 29, 1992, EPRI stated that to address the NRC concerns about the loss of a Class 1E dc or vital ac bus and at the same time clarify the industry concerns about plant availability, it would revise Sections 7.2.1 and 7.2.2 in Chapter 11 of the passive Requirements Document. The following changes were made to the requirements in those sections by Revision 4 of the passive Requirements Document:

> Section 7.2.1:
>
> The Class 1E dc and vital ac power supply system shall be designed with sufficient redundancy to ensure that the loss of either a dc or a vital ac bus will not result in a plant transient and simultaneous loss of single failure protection in the systems needed to respond to the event.
>
> Section 7.2.2:
>
> The dc and low voltage ac power supply systems shall be designed with sufficient redundancy to ensure that:
>
> - The loss of Class 1E dc battery or battery-charger or vital ac bus will not result in a plant transient such as a trip, scram, isolation or safety systems initiation.
>
> - Battery equalization or any other maintenance activity on a battery, battery charger or inverter may be performed off-line without affecting continuous plant operation at 100 percent power due to Technical Specifications (TS) limitations for at least seven days.

Corresponding changes were also made to the rationale related to the above requirements. The staff concludes that the above changes, with the exception of the second bullet in Section 7.2.2, agree with its position on this matter and are, therefore, acceptable. Therefore, the DSER open issue is closed. The second bullet in Section 7.2.2 relates to TS issues that the staff will not address until it reviews an individual application for FDA/DC.

7.2.2 Battery Sizing Criteria Specified for the 72-Hour Coping Capability

Sections 7.2.3 and 7.3.2.4 of Chapter 11 provide the loading and sizing criteria for the batteries in the dc power supply system. They provide two sizing criteria, one based on supplying power to the loads for at least 2 hours and the other based on supplying power to selected safety loads for at least 72 hours. The staff indicated to EPRI that both of these criteria should stipulate that the loads supplied by the batteries and the load profiles developed to size the batteries should include the loads and actions necessary to recover from the events specified in the criteria. This is necessary so that it is clear that a designer must size the batteries not only to maintain power to the steady-state loads, but also to supply the short-time

and transient loads embodied in the recovery actions, such as breaker switching and standby power source starting. EPRI gave the following response in its January 10, 1992, letter:

> The recovery actions that will take place at the end of the 72 hour period are expected to require little or no dc power as they will consist essentially of connecting the battery chargers to transportable ac power generators. The battery charger will then provide the dc power required to recharge the batteries and operate the control systems needed to perform additional recovery actions. Furthermore, it is expected that manual switchgear operation, i.e., with no control power required, will be possible. Nevertheless, in response to the staff's comment, a paragraph will be added to the requirement of Chapter 11, Volume III, Section 7.2.3, as follows:
>
> "The load profiles used to size the batteries shall include the short-time and transient loads, e.g., breaker switching loads or standby power source starting, expected to be required at the end of the corresponding coping period, e.g., 2 hours or 72 hours, to permit recharging the batteries and performing other appropriate recovery actions."

The rationale portion of Section 7.3.2.4 of Chapter 11 deals with the sizing of the batteries to ensure the 72-hour coping ability. It originally stated that the sizing of these batteries would be governed mainly by the power requirements for postaccident monitoring and emergency lighting. It further stated that these batteries would also be used to supply power to other loads such as Class 1E valve operators and solenoids, as operation of these loads during the coping period should be possible and the effect on battery sizing would be negligible because of the loads' intermittent duty. The effect of intermittent transient loads such as valve operators, circuit breaker switching, and diesel generator starting should not be underestimated. Although these loads are of short duration, they often have a high-current demand. When they are needed to recover from an event near the end of 72 hours, this high current demand could result in the dc voltage being pulled below the battery final voltage or the rated voltage of the loads, whichever is limiting. The result might be that a much larger battery would be needed to maintain adequate voltage than that needed to only supply the steady-state loads such as postaccident monitoring and emergency lighting. The staff, therefore, recommended that EPRI modify the rationale portion of Section 7.3.2.4 to indicate that the battery must be capable of supplying the short-duration loads that are necessary for recovery from the most limiting event and eliminate the reference to their negligible effect on battery sizing. In its January 10, 1992, letter EPRI provided the following response:

> In response to the staff's comment, the second and third sentences of the rationale of Chapter 11, Section 7.3.2.4, Volume III will be revised as follows:
>
> "The sizing of these batteries will be governed mainly by the power requirements for post-accident monitoring and emergency lighting during the 72-hour coping period and for the recovery actions, e.g., switchgear operation, needed at the end of the coping period. It is expected that these batteries will also be used to supply power to

other loads such as Class 1E valve operators and solenoids, etc., as operation of these loads during the coping period should remain possible."

In the DSER for Chapter 11, the staff concluded that the changes proposed by EPRI for these two items were consistent with its recommendation and were, therefore, acceptable. It identified this as a confirmatory issue.

These changes have been incorporated in Revision 3; therefore, this issue is closed.

7.2.3 Load Shedding for 72-Hour Battery Endurance

Section 7.2.4 of Chapter 11 states that the dc and low-voltage ac power supply system components required to cope with a loss of all ac power sources will be designed to operate for at least 72 hours without forced cooling or ambient cooling. In parentheses, EPRI had originally noted that it would assume that all electrical loads not required to cope with the loss of ac sources had been deenergized during this period. The staff indicated to EPRI that the parenthetical sentence should be deleted because it was not consistent with the requirement in Section 7.2.3 that the batteries be sized for at least 72 hours following a licensing-basis event (which includes the loss of all ac sources) without manual load shedding or load management program. The same parenthetical statement was made in the evolutionary Requirements Document; however, those requirements specifically assumed that manual load shedding and load management programs would be used. In a letter dated January 10, 1992, EPRI gave the following response:

> It is acceptable to assume that no heat will be generated by electrical loads not required to cope with an extended loss of ac power since, as indicated by Requirements 7.2.3 and 7.3.2.4, only a relatively small number of loads are expected to remain available beyond 2 hours. Low battery voltage at the end of two hours will cause breakers to trip and loads to be shed. In order, however, to eliminate any possible misinterpretation the parenthetical expression of Chapter 11, Volume III, Section 7.2.4 will be revised as follows: "In meeting this requirement, credit may be taken for the reduction of heat loads resulting from the loss of power supply to electrical loads not required to cope with the event."

In the DSER for Chapter 11, the staff concluded that this change by itself was not acceptable. Low battery voltage at the end of 2 hours will <u>not</u> cause breakers to trip unless undervoltage (UV) trip devices are installed in the circuit breakers. UV trip devices are not routinely installed in circuit breakers but are available as a special feature. The staff had not seen a requirement for the devices in the passive Requirements Document so it indicated that it should not be assumed that they will be installed in the circuit breakers. If they are intended to be installed, consideration must be given to how they might create inadvertent breaker trips, such as during transient voltage dips. For loads installed on batteries with only 2 hours' endurance, for heat calculation purposes, it could be assumed the loads are deenergized at some time beyond 2 hours whether their breakers trip or not, since there eventually would not be any battery source voltage. However, the staff stated that protecting the loads from the low voltage would also have to

be addressed if the low voltage could cause equipment damage or unintended operations or indications. The staff also stated that shedding of unnecessary loads from batteries with 72 hours' endurance must also be treated and indicated that EPRI should address all these staff comments. The staff further stated that if EPRI wished to take credit for reduction of loads not used for the 72-hour endurance, either for the calculation of heat loads or for the calculation of battery endurance, EPRI should describe how this will be accomplished and should provide the associated requirements in the passive Requirements Document. This was identified as an open issue.

In its letter dated September 29, 1992, EPRI responded to these issues. With regard to the shedding of unnecessary loads from batteries with 72 hours' endurance, EPRI stated that the 72-hour dc buses of the passive plants will not need automatic or administrative shedding because their batteries will be sized for 72 hours' endurance without shedding. This clarifies EPRI's intent. However, Section 7.2.3 of Chapter 11 still only prohibits manual load shedding and not automatic load shedding on the 72-hour batteries, although the rationale does state that the need for automatic load-shedding schemes should be avoided. Therefore, if automatic load-shedding schemes are used in the passive designs, the staff will review the details of those schemes during its review of an individual application for FDA/DC.

With regard to the effects of low battery voltage, EPRI provided the following modifications to Chapter 11, Section 7.2.7, in Revision 4 of the passive Requirements Document:

> . . . . steady state and switching transients. If some of the plant batteries are sized for less than 72 hours endurance (for instance 2 or 24 hours), plant the designer should also analyze the effects of operating the loads under the low voltage conditions that would result beyond the endurance period of the battery. If the low voltage could cause unacceptable equipment damage or unintended operations, undervoltage disconnecting units with adequate filtering and discriminating features should be incorporated.
>
> Rationale:
>
> . . . . under all expected conditions. The provisions with regard to the batteries with less than 72 hours endurance time are not required for the 72-hour batteries since, at the end of that period, portable ac generators to recharge the batteries are expected to be available at the site.

The staff concludes that the above changes satisfactorily address its comments regarding the effects of low battery voltage, with the exception of unintended or anomalous indications, and are, therefore, acceptable. The staff will address the issue of unintended or anomalous instrumentation indications during its review of an individual application for FDA/DC.

With regard to the reduction of load for heat calculation purposes on batteries that are sized for less than 72 hours' endurance, EPRI submitted the following modification to Chapter 11, Section 7.2.4, in Revision 4 of the passive Requirements Document:

. . . . or ambient cooling (HVAC). In meeting this requirement, credit can be taken for the reduction of load associated to the most limiting discharge profile considered in the design of the plant batteries.

This modification is acceptable if "the most limiting discharge profile" is assumed to be the most limiting in terms of heat generated rather than battery endurance. Battery derating factors such as those due to temperature, battery age, and load margins are limiting for endurance calculations but not for heat calculation purposes. A separate discharge profile would, therefore, have to be created for heat calculation purposes. The staff will ensure that the correct discharge profiles are used when it reviews an individual application for FDA/DC.

7.2.4 Backup Battery and Battery Charger

Section 7.2.6 of Chapter 11 requires that during normal plant operation, the batteries and battery chargers generally be continuously connected to the dc distribution buses. The rationale given is to ensure system readiness; specifically, batteries must remain fully charged during normal operation and automatically assume the system loads on loss of ac power or failure of battery chargers. Figure 11.7-1 in Chapter 11, however, shows a backup battery and battery charger that is normally not connected to any dc system bus. Therefore, the staff indicated to EPRI that the Section 7.2.6 requirement should be revised to include the configuration for the backup battery and battery charger shown in Figure 11.7-1. In the January 10, 1992, letter, EPRI responded that the following sentence would be added at the end of Section 7.2.6 of Chapter 11: "The requirement does not apply to batteries or battery chargers exclusively provided for backup purposes."

Section 7.3.1.4 of Chapter 11 requires that provision be made to permit connecting each dc bus to a standby, backup dc source (i.e., a combination of a battery and a battery charger). It further requires that this backup source have sufficient capacity to permit continuous plant operation at 100-percent power in case of failure or unavailability of the normal battery or battery charger. The staff asked EPRI whether the backup source would be needed in addition to the normal dc sources to allow the passive safety systems to operate for 72 hours following an event including station blackout. If it would be needed, TSs would not allow the backup source to be considered as a replacement for a normal source, and operation in the replacement mode would be limited. If the backup dc source would not be needed for the passive safety systems to operate 72 hours following an event, then, in order for TSs to permit continuous plant operation at 100-percent power, the capacity and capability of the backup battery and charger must at least be equal to that of the battery and charger it replaces. The use of the backup battery and charger must also not reduce the separation found when the systems are operating on their normal dc power sources. In the January 10, 1992, letter, EPRI gave the following response:

> It is the intent of Requirement 7.3.1.4 that the backup battery will not be needed to ensure operation of the passive safety systems for 72 hours following a licensing basis event. It is also the intent of this requirement that the capacity and capability of the backup battery and charger will be at least equal to that of the battery or

charger it replaces. In addition, separation requirements for normal operation should be maintained when a division or channel of the Class 1E dc system is fed from the backup dc source. In order to prevent any misinterpretation of the requirement, the following will be added to the end of the existing requirement of Chapter 11, Section 7.3.1.4, Volume III:

". . . or unavailability of the normal battery or battery charger, i.e., the capacity and capability of the backup battery and charger shall be at least equal to that of the battery or charger it replaces. This backup source shall not be needed to meet the capacity requirement of Section 7.2.3. Separation between redundant safety systems shall be maintained when a system is fed from the backup dc source."

In the DSER for Chapter 11, the staff concluded that the responses to these two items were consistent with its recommendations and were, therefore, acceptable. However, the staff stated that EPRI should amend the proposed changes to the passive Requirements Document to also require that provisions be made to keep any backup batteries used in the design in a ready state of charge, if they are normally not on line, so that they could be made available as a replacement for the normal batteries in a relatively short time. The configuration of the backup battery and battery charger shown in Figure 11.7-1 in Chapter 11 meets this additional requirement, but a corresponding requirement was not explicitly contained in the text of the passive Requirements Document. This was identified as an open issue.

In its letter dated September 29, 1992, EPRI responded that the intent of the passive Requirements Document is to provide such a capability and identified a change to Chapter 11, Section 7.3.1.4, to clarify this intent. This change has been incorporated by Revision 4 into the passive Requirements Document and satisfactorily resolves this issue; therefore, this open issue is closed.

7.2.5 Physical and Electrical Separation of the Safety-Grade DC and Low-Voltage Vital AC Power Supply System

Section 7.3.2.1 of Chapter 11 required that the safety portions of the dc and low-voltage vital ac power systems be divided into a minimum of two separate divisions and the components of each division be physically separated and electrically independent of the components of the other division. This implied that components within one of the two divisions need not be physically separate or electrically independent. The staff told EPRI that this conflicted with the Section 7.3.2.2 requirement that the arrangement of the safety dc and low-voltage ac power supply systems be consistent with the requirements for the number and independence of reactor protection channels and reactor trip logic specified in Chapter 10. Section 7.3.2.2 further stated that "specifically, this arrangement shall include four separate sets of Class 1E, dc and low-voltage vital ac circuits to supply power to RPS [reactor protection system] channels." Therefore, the safety portions of the dc and low-voltage vital ac power systems should be divided into a minimum of four separate divisions consistent with four RPS channels. Referring to Figure 11.7-1 in Chapter 11, the power sources to RPS Channels A, B, C, and D should be physically separate and electrically independent of each other. This separation should be maintained from the RPS distribution buses up

through to their associated batteries and chargers. Also, the Division I dc power source shown in Figure 11.7-1 should be physically separate from and electrically independent of the Division II dc power source. The staff asked EPRI to clarify Section 7.3.2.1 to indicate that this is the case. In the January 10, 1992, letter, EPRI gave the following response:

> The requirement for separation between the Division I and Division II power sources shown in Figure 11.7-1 is specified in Section 7.3.2.1. As recommended in the Question, Sections 7.3.2.2 and 7.3.2.5 require that each RPS channel be fed from a Class 1E power source, i.e., a battery, battery charger, dc bus and inverter, that is separate and independent from any other power source. Two passive safety divisions are provided for the passive plant (see responses to Questions 435.38 and 435.41). This is consistent with Section 7.3.2.1. The number of RPS channels is greater than the number of divisions in order to implement a coincident logic scheme to generate reactor trip and accident signals, not in order to provide N-2 (or better) level of protection against possible failures of safety functions. The purpose of the separation requirements for the RPS channels is to obtain the full benefits of the coincident logic scheme.

To ensure that Section 7.3.2.1 is not interpreted as implying that components within one of the two divisions need not be separated and independent, this requirement was modified in Revision 4 as follows: "Additional independence and redundancy requirements for the dc and low-voltage ac systems are included in Sections 7.3.2.2 and 7.3.2.5."

Section 7.3.2.5 of Chapter 11 requires that each reactor protection and safety systems actuation channel normally be powered from a dedicated Class 1E dc source that is normally independent of the dc sources of the other channels and of other dc sources. It further states that each source will include a battery and a battery charger and will be capable of supplying power to its associated channel for a minimum period of 2 hours. Note 2 originally in this section stated that this requirement did not apply to designs with N-2 safety divisions since, in this case, N-1 divisions of safety systems were expected to remain available even in the event of a failure of a battery simultaneously powering a reactor protection channel and a division of safety systems. The note was ambiguous because it did not explicitly identify which of the Section 7.3.2.5 requirements did not apply, nor was it clear which N-2 safety divisions (RPS channels, safety system actuation channels, or passive safety systems divisions) were being addressed. As a result, it could be interpreted to allow more exceptions to the requirements than the staff believed was its intent. Apparently, the intent of the note was to allow an RPS channel and a safety systems actuation channel to be powered from a common Class 1E dc source that would also be powering a division of passive safety systems instrumentation and control devices if there were more than two redundant passive safety systems divisions. The requirement that each RPS channel have its own power source that is separate from the power sources of the other RPS channels would remain, and the requirement that each safety systems actuation channel have a power source (shared with one RPS channel) that is separate from the power sources of the other safety systems actuation channels would also remain. However, the note as written could be interpreted to allow redundant RPS channels to share power sources and redundant safety systems

actuation channels to share power sources, or to apply to N-2 divisions of RPS channels. It also was not clear why Note 2 was needed, since only two passive safety systems divisions had been specified for the passive ALWR. Therefore, the staff recommended to EPRI either that the note be deleted completely or that it be made explicitly clear what exception to the requirements are allowed and which N-2 safety divisions are being addressed. The rationale for this requirement should also contain the rationale for both Notes 1 and 2. EPRI gave the following response to this item in the January 10, 1992, letter:

> The staff's interpretation of the note is correct since Requirements 2.3.5 and 7.3.2.2 essentially prohibit that two redundant safety system actuation channels share the same power source and, similarly, that two RPS channels be fed from the same source. In any event, Note No. 2 of Section 7.3.2.5, Volume III will be deleted since, as observed by the staff, only two divisions of passive safety systems will be provided for the passive plant. A rationale for Note No. 1 of Chapter 11 Section 7.3.2.5, Volume III will be included as follows:
>
> > Provisions for cross-connections between dc sources within a division are permitted as they do not compromise divisional separation and may provide operators with additional flexibility in utilizing the plant's installed battery capacity.

These changes were included in Revision 3 of the passive Requirements Document.

In the DSER for Chapter 11, the staff accepted EPRI's rationale proposed above for allowing cross-connections between dc sources within a division with the understanding that the cross-connections are only allowed between dc sources that are not related (e.g., between the Division I dc power source and the Channel A dc power source).

In the DSER, the staff also stated that it was clear from EPRI's response to these two items that the intent is that the four dc and ac power supplies to the four RPS and safety features actuation channels, and the two divisional dc power supplies, will be normally separate and independent of each other as shown in Figure 11.7-1 in Chapter 11. Figure 11.7-1, however, also showed a common backup ac power source (regulating transformer) between the Channel A inverter and the Channel B inverter. A common backup power source was also shown between the Channel C and the Channel D inverter. The inverters were labeled as having a static switch, and Section 7.4.4.2 in Chapter 11 indicated that these switches will be provided for automatic transfer of the loads to the backup ac power source. The staff stated in its DSER that such an automatic transfer scheme of independent channels to a common backup source could jeopardize the independence of the channels. It stated that the transfer scheme should either be made totally manual utilizing redundant isolation devices (combination of two open switches or circuit breakers) between the channels at all times, or separate backup ac power sources should be provided for each channel. This was identified as an open issue.

In a letter dated March 31, 1993, EPRI submitted a markup of Section 7.4.4.2 and Figure 11.7-1 in Chapter 11 of the passive Requirements Document to resolve this issue. The following sentence has been added to Section 7.4.4.2,

"Each dc division shall be provided with a dedicated regulating transformer as a backup source." In the Figure 11.7-1 markup, a separate regulating transformer connected to each inverter/static switch combination is shown. These changes agree with the staff recommendations and are acceptable. Therefore, this open issue is closed.

The staff stated in its DSER that final review of separation criteria, however, would require more detailed information on the circuits involved and how the separation would be maintained. That information is typically only available in a specific design. Therefore, the staff stated that it would evaluate the details of the electrical separation during its review of an individual application for FDA/DC. The staff stated that it would also confirm that the changes committed to by EPRI in the above two items were acceptably incorporated into Chapter 11 of the passive Requirements Document.

Following its review of the staff's DSER, EPRI provided additional clarification on the electrical separation requirements in its letter dated September 29, 1992. EPRI indicated that the intent of the requirements in the passive Requirements Document was not to necessarily require that the four power supplies to the four RPS channels be physically separated (independent) from each other, as stated in the DSER, but rather only to require that a minimum of two physically separated divisions be provided. The two power supplies located within a single division would, therefore, not necessarily have to be separated from each other. In further discussions with the staff EPRI also stated that the separation (independence) of the power supplies was required to be consistent with the separation (independence) of the reactor protection channels and reactor trip logic specified in Chapter 10. This requirement for consistency is specified in Section 7.3.2.2 of Chapter 11. A lack of separation or independence between power supplies would be allowed, therefore, only if the same lack of independence were allowed in the instrumentation channels they power. EPRI also indicated that the requirements in Chapter 10 did not specifically require that four RPS channels be provided, but rather prescribed only functional requirements for the RPS. EPRI, therefore, eliminated the reference to four channels in Section 7.3.2.2 of Chapter 11. The staff has verified that the changes related to Section 7.3.2.2 have been incorporated into the passive Requirements Document. Therefore, this confirmatory issue is closed.

The staff concludes that the overriding requirement related to this issue is that specified in Section 7.3.2.2 of Chapter 11 that the number and independence of the power supplies be consistent with the requirements for the reactor protection channels and reactor trip logic specified in Chapter 10. The staff's basis for this conclusion is that the requirements specified for the power supplies should be no less than the requirements specified for the channels they power, in order to realize the full benefit of those functional requirements. In Revision 5, EPRI changed the requirement to include additional details of design for physical separation. The staff has reviewed Figure 11.7-3 and concludes that, assuming the hazards (missels, fire, high and moderate energy pipe breaks effects, environmental qualification concerns, etc.) are properly addressed in the specific applications for FDA/DC, EPRI's treatment of separation appears reasonable and is consistent with regulatory requirements. The staff will evaluate this aspect of the design during its review of an individual application for FDA/DC.

7.2.6 Non-Safety DC Power Supplies for Switchyard Circuit Protection and Control Equipment

Section 7.3.3.3 of Chapter 11 requires that the non-safety 125-V dc power supply system include redundant power supply buses, batteries, and battery chargers and that redundant chargers be fed from separate ac buses. In the rationale portion of this requirement, EPRI originally stated that redundant, non-safety dc power supplies were required for operation of some redundant non-safety loads, including switchyard circuit protection and control equipment, step-up transformer auxiliary equipment, and switchgear controls. Figure 11.7-2 in Chapter 11 shows only a single non-safety 125-V dc power supply system. The staff assumed that this non-safety dc power system would be used to support the non-safety ac standby power source and distribution systems. If this was the case, EPRI should specify additional non-safety batteries and dc power supplies to provide power for switchyard circuit protection and control equipment that are separate from those that supply the non-safety ac standby power sources and distribution systems, consistent with the requirements of Section 3.6.3 of Chapter 11. The staff, therefore, asked that EPRI clarify Section 7.3.3.3 and Figure 11.7-2 to indicate that this is the case. EPRI responded as follows in the January 10, 1992, letter:

> As indicated in Section 3.6.3, the protection and control devices for switchyard equipment will be fed from redundant batteries separated from those supplying dc power to other non-safety loads. Section 7.3.3.3 will be clarified as indicated below. Figure 11.7-2, however, is for illustration only and was not intended to show the switchyard batteries. Section 7.3.3.3 will be revised as follows:
>
> Requirement:
>
> The following sentence will be added at the end of the existing requirement:
>
> "Two redundant, dedicated batteries shall be provided to supply power to switchyard loads in accordance with Requirement 3.6.3."
>
> Rationale:
>
> The first sentence will be revised as follows:
>
> ". . . redundant non-safety dc power supplies are required for operation of some redundant non-safety loads such as those associated with the standby power sources and their distribution systems. Redundant power supplies are also required to supply power to redundant switchyard loads such as protection and control devices, step-up transformer auxiliary equipment, switchgear controls, etc."

In the DSER for Chapter 11, the staff stated that EPRI's proposed revisions acceptably clarified that the non-safety redundant switchyard batteries will be separate from the non-safety redundant plant batteries and identified this as a confirmatory issue.

These changes were made in Revision 3; therefore, this issue is closed.

7.2.7 Battery Capacity Margin

Section 7.4.2.2 of Chapter 11 originally stated that the batteries would be provided with a minimum capacity margin of 10 percent to allow for future load growth, and this margin would be provided in addition to the capacity margins required to compensate for temperature effects, aging, and initial capacity. IEEE 485 calls this "design margin" and indicates that it is used to allow for unforeseen additions to the dc system or less-than-optimum operating conditions of the battery (due to improper maintenance, recent discharge, or ambient temperatures lower than anticipated), or both. Because this margin is used for more than just future load growth, and because the load growth in a passive ALWR is likely to be greater than in past plants because it is a new concept with more unknowns, the staff recommended to EPRI that a minimum margin of 20 percent be specified. In the January 10, 1992, letter, EPRI gave the following response:

> The battery capacity requirements for the passive plant are considered very conservative. In addition to requiring that no shedding be assumed and that margins be included for temperature effects and aging, an additional spare battery is also specified which will provide additional protection against failures and unexpected operating conditions. Furthermore, the number of unanticipated loads to be added to the original batteries should be minimal in the case of a standardized certified plant. Accordingly, a capacity margin of 10 to 15 percent is considered adequate. Section 7.4.2.2 will be revised as follows: "The batteries shall be provided with a capacity margin of 10 to 15 percent to allow for future load growth."

If the additional spare battery that EPRI refers to in its response will be used to provide margin for additional load growth in the future, then at the time the loads are added, it could no longer be considered a spare battery, and unavailability of the battery would have to be limited by technical specifications. Regardless, the question of insufficient load-growth margin is primarily an economic concern that could result in the need for future additional battery capacity. There are no regulatory requirements governing load-growth margin in safety-related batteries. EPRI has agreed to provide a load-growth margin of 10 to 15 percent in the batteries. Therefore, the staff concluded in the DSER for Chapter 11 that EPRI's response was acceptable and identified this as a confirmatory issue.

The changes has been made in Revision 3 of the passive Requirements Document; therefore, this issue is closed.

7.2.8 Battery and Battery Charger Instrumentation and Alarms

Section 7.6.2 of Chapter 11 identifies the following instrumentation and alarms that are required to be installed to monitor the status of the battery and battery charger in the passive ALWRs:

- Indications:
  - voltmeter for dc bus voltage

- ammeter(s) for battery current (charge and discharge)
- ammeter for primary and backup charger output current
- voltmeter for battery charger output voltage

• Alarms:

- high and low battery voltage alarm
- high and low dc bus voltage alarm
- battery charger undervoltage
- battery discharge alarm
- breaker open alarms (battery and battery charger input and output breakers as a minimum)
- high impedance ground fault detection

In addition to these alarms and indications, the staff recommended in the DSER for Chapter 11 that an ampere-hour discharge meter be installed for monitoring the safety-related batteries that are required to provide the 72-hour endurance capability. The meters would be very useful to the operators during an event involving loss of ac power to more easily determine the amount of power used and remaining capacity available from these batteries. Because of the relatively long endurance required of the 72-hour batteries, more time will be available to the operator to monitor this parameter. A knowledge of the remaining battery capacity and approximate time to battery final voltage will help the operators plan the recovery actions over the course of such an event. An ampere-hour meter would also be helpful in conducting discharge-test surveillance on these batteries. This was identified as an open issue.

In its response dated September 29, 1992, EPRI stated that the instrumentation that has been specified in the passive Requirements Document is based on current practice and regulations; the additional feature proposed by the NRC should be considered by the designers during their design process as appropriate, depending on the functions and tasks analysis required in Chapter 10, Section 3. EPRI stated that this issue should, therefore, be a vendor-specific item. The staff agrees and will pursue this issue during its review of an individual application for FDA/DC. Therefore, this open issue is closed.

7.2.9 Lack of Requirement for Electric Protective Assemblies in RPS Power for BWRs

Section 5.3.5.3 in Chapter 4 of the passive Requirements Document states that the scram pilot solenoid valves will be designed for continuous operation at the minimum and maximum voltages and frequencies required by the reactor protection system (RPS). In the rationale portion of this section, EPRI states that this design will avoid overheating and consequent damage to valve materials observed in earlier plants. Regarding the evolutionary plant, which has an identical requirement, the staff asked EPRI whether this requirement

was meant to replace that for the electric protective assemblies (EPAs) that were used in earlier BWR designs to prevent an overvoltage, undervoltage, or underfrequency condition from failing the scram pilot solenoid valves in a non-fail-safe state.

In its response regarding the evolutionary plant, EPRI described a failure mode involving the scram discharge volume that had already been eliminated for the advanced boiling-water-reactor (ABWR) design but was not the original reason for adding the EPAs. The failure mode identified by EPRI was the potential, in previous BWR designs, for a low-voltage condition to allow some scram discharge valves to open and fill the scram discharge volume without a scram of all control rods occurring. The original basis for requiring the EPAs, however, was the concern that an overvoltage, undervoltage, or underfrequency condition would cause excessive current or vibration of the scram pilot solenoids resulting in overheating and eventual lockup of the solenoids. Eliminating the scram discharge volume will eliminate the failure mode identified by EPRI but not other failure modes. In a subsequent discussion with the staff, EPRI stated that the intent of this section of the Requirements Document was to preclude the overheating and binding of the solenoids by requiring that they be designed to operate continuously over the full range of voltages and frequencies that could be put out by the RPS power supplies. EPRI intends to demonstrate this through a failure modes analysis of the power supplies. The staff expressed its skepticism that it was possible to demonstrate that there were no failure modes of the RPS power supplies that would result in a non-fail-safe failure of the scram pilot solenoids.

By letter dated February 11, 1992, EPRI continued to maintain that the function of the EPA would not be needed because of the elimination of the scram discharge volume. As stated above, the staff maintained that eliminating the scram discharge volume would not eliminate the failure mode that was the original basis for the need for the EPA function. Furthermore, EPRI's approach of specifying requirements for the power supply and scram pilot solenoids, while needed, would not eliminate the full range of postulated failures that resulted in the need for the EPAs. The staff, therefore, pursued the need for the EPA function with General Electric (GE) during the ABWR review. On the basis of discussions with GE, it appeared that GE had included the EPA function in the ABWR design. The staff, therefore, concluded in the DSER for Chapter 11 that because, to the extent the staff could determine at that time, the ABWR design and the passive BWR design would be virtually identical in this area, the passive BWR design should also incorporate the use of the EPA function. EPRI should, therefore, include a requirement in the passive Requirements Document to this effect. This was identified as an open issue.

In its response dated September 29, 1992, EPRI indicated that for the ABWR design GE had developed a solid analysis to demonstrate that the postulated failure mode of the RPS power supplies that would lead to the need for the EPAs was very unlikely, but agreed to include them in the ABWR design after it reviewed the analysis further because during discussions with the NRC staff a failure mechanism that would lead to that postulated failure mode was identified. EPRI stated, however, that GE maintains that the EPAs are not considered necessary features for the ABWR and simplified (SBWR) and that GE had included them in the design to continue with the certification process.

EPRI also stated that operating experience pertaining to RPS power supplies in BWRs supports the initial analysis of GE and identified the few events involving the types of failures that the EPAs would protect against. EPRI concluded from the data that the postulated failure mode leading to the need for EPAs is very unlikely if the RPS loads are powered from regulated power supplies such as the static inverters.

EPRI further indicated that when the EPAs were first required, the BWRs had only two scram features (the scram ac solenoids and the backup scram dc solenoids), but the ABWR and SBWR now have four diverse features for rod insertion capability (the scram ac solenoids, the backup scram dc solenoids, the alternate rod insertion (ARI) dc solenoids, and the fine motion control rod drives (FMCRDs)). In a response to this issue dated April 14, 1993, EPRI also stated that the scram ac solenoids are powered from Class 1E 120-V ac distribution panels, the backup scram dc solenoids are powered from Class 1E dc buses, the ARI dc solenoids are powered from non-Class 1E 125-V dc buses, and the FMCRDs are powered from 480-V ac motor control centers. The EPAs would only protect the primary scram capability of the scram ac solenoids. The staff notes that the passive Requirements Document now also requires capability for automatic actuation of the standby liquid control system.

The staff concludes that because the scram ac solenoids are the primary means of scramming the reactor and provide immediate insertion of all rods, a system for detecting the occurrence of undervoltage, overvoltage, and underfrequency on the power sources to those solenoids should continue to be used in the advanced BWR designs to preclude the occurrence of a non-fail-safe undetectable failure of the solenoids. However, because of the additional diverse means available to scram the reactor at the advanced plants and the amount of time generally necessary to fail the solenoids, an alarm system capable of detecting the above occurrences would be acceptable. The alarm system should be independent of the power supplies they monitor and should alarm in the main control room when an out-of-range condition occurs. The staff will pursue the implementation of such an alarm system at the advanced BWRs during its review of an individual application for FDA/DC. Therefore, this open issue is closed.

7.2.10 Electrical Equipment in Vital Areas

The portions of the dc and low-voltage ac power supply systems that support equipment that is required for safe shutdown under design-basis-accident assumptions are designated Class 1E and, hence, seismic Category I. On the basis of SRP Section 13.6, "Physical Security," and its Review Guideline 17, the staff considers that these portions of the power supply systems are equipment of the type required by 10 CFR Part 73 and by Section 5.2.1.1 of Chapter 9 of the passive Requirements Document to be located within one or more vital areas. In addition to the security equipment backup power supply as discussed in Section 5.2.12 of Chapter 9 of the passive Requirements Document, the staff concludes that only the seismic Category I portions of the plant power supplies are required to be in a vital area. However, as discussed in Appendix B to Chapter 1 of this report, the Commission has not defined the treatment of non-safety systems for passive plant designs. Depending on the resolution of this issue, additional ac power equipment may need to be protected as vital equipment.

7.2.11 Uninterruptible Power Supply for Security Equipment

NUREG-0908, "Acceptance Criteria for the Evaluation of Nuclear Power Reactor Security Plans," states that under an acceptable security program, the alarm stations would typically be provided with a source of emergency power capable of supplying power for all required security functions. American National Standards Institute/American Nuclear Society (ANSI/ANS) 3.3-1988 specifies that security intrusion detection aids should be supplied with uninterruptible power.

Revision 2 to the passive Requirements Document deleted security systems from the list of non-safety low-voltage ac systems to be provided uninterruptible power from the plant's common power supply systems, because requirements for a separate, dedicated backup power source for security systems were added to Section 5.2.12 of Chapter 9 of the passive Requirements Document. The staff found that power supply acceptable in its review of Chapter 9 of the passive Requirements Document.

8 NORMAL AND EMERGENCY LIGHTING

8.1 Function and Description

Section 8 of Chapter 11 of the passive Requirements Document provides design criteria for all onsite systems that will provide artificial illumination for rooms, spaces, and outdoor areas of the plant. These systems will include a normal station lighting system, a security lighting system, and an emergency lighting system. Section 8.2.1 of Chapter 11 states that illumination will be provided for each area of the plant in accordance with the guidelines of the IES Lighting Handbook, published by the Illumination Engineering Society, at the time the plant is designed.

EPRI states that the normal station lighting system will be used to provide normal illumination under all plant operating, maintenance, and test conditions. Section 8.3.3 of Chapter 11 states that the normal lighting system is considered by EPRI to be part of the plant's permanent non-safety systems and, as such, is required by EPRI to be energized as long as power from the normal power source or a standby non-safety source (combustion turbine or diesel generators) is available.

The security lighting system will provide illumination required to monitor isolation zones and all outdoor areas within the plant's protected perimeters, under normal plant conditions as well as on loss of all ac power. Revision 2 to Section 8.4.1 of Chapter 11 requires uninterruptible power for those portions of the security lighting that are essential to plant protection following interruption of normal power. The balance of the security lighting system will be powered from the dedicated security generator specified in Section 5.2.12.4 of Chapter 9. Section 8.4.2 of Chapter 11, as modified by EPRI's letter dated January 24, 1992, specifies that the security lighting illumination level will be designed to be compatible with security monitoring equipment and to provide a minimum illumination of 1 foot-candle in selected areas, such as the isolation zone, and a minimum of 0.2 foot-candle elsewhere in the protected area.

The emergency lighting system will be used to provide acceptable levels of illumination throughout the station and in areas in which emergency operations will be performed, such as control rooms, battery rooms, and the containment, on loss of the normal lighting system. Section 8.5.1 of Chapter 11 specifies that the emergency lighting system will provide illumination units of at least 10 foot-candles in those areas of the plant in which emergency operations will be performed that could require the reading of printed or written material or the reading of scales and legends. In other areas of the plant, EPRI requires the emergency lighting to be able to achieve a minimum illumination level of 2 foot-candles.

Section 8.5.2 of Chapter 11 states that emergency lighting in the main control room will be powered from the safety uninterruptible (ac/dc) power supply. Outside the main control room, emergency lighting will be provided by dc self-contained, battery-operated lighting units. These units will be sized to provide at least 8 hours of operation at rated load.

The necessity to enter high-radiation areas on a frequent basis to relamp burned-out lighting fixtures can be a source of unnecessary personal radiation exposures. In an effort to reduce the doses associated with relamping and maintenance on lighting fixtures, the plant designer will locate lighting fixtures in low-radiation zones, use long-life lighting, and provide easy access to lighting fixtures. These features meet the intent of RG 8.8 to maintain occupational doses as low as is reasonably achievable (ALARA) and are acceptable.

8.2 Evaluation

8.2.1 Availability of the Normal and Emergency Lighting Following Design-Basis Events

Section 8 of Chapter 11 of the passive Requirements Document specifies the lighting systems to be used in the ALWR passive plant designs. It calls for a normal and an emergency lighting system. The normal lighting system will provide illumination for all plant areas during normal operation. Section 8.3.3 of Chapter 11 requires that the normal lighting system be considered part of the plant permanent non-safety system, and, as such, it must be energized as long as power from the normal power sources or a standby source is available. The emergency lighting system will be used to light vital areas of the plant when the normal lighting system is lost. Section 8.5.2 of Chapter 11 requires that emergency lighting outside the main control room be provided by dc self-contained, battery-operated lighting units. Section 8.5.3 of Chapter 11 specifies that the battery life of the battery-operated lighting units will be at least 8 hours. Therefore, all lighting inside the plant but outside the main control room will be provided by either the non-safety ac distribution system or by self-contained, battery-operated lighting units good for only 8 hours. The staff asked EPRI to justify the loss of all plant lighting outside the main control room only 8 hours into the 72-hour capability of the passive safety systems for an event that results in the loss of the non-safety ac distribution systems. Such an event might be caused by a severe seismic disturbance or could also result from an SBO. EPRI provided the following response in its January 10, 1992, letter:

> As indicated in Section 2.3.2.9 of Volume III, Chapter 1, no operator action is required to maintain plant safety for at least 72 hours following a licensing basis event including one which results in a complete loss of all ac power sources. Certainly, no operator action outside the main control room will be necessary. Sufficient lighting must however be provided to ensure personnel safety and permit plant evacuation. A lighting period of 2 hours is typically considered appropriate for this purpose.
>
> In addition, emergency lighting is also provided (see Sections 8.5.1 and 8.5.2) to permit control action to be taken outside the control room during events which require the evacuation of the control room, as a result of a fire for example. Although the normal lighting circuits and their power sources are expected to remain available, at least in part, during such events, self-contained battery-operated lighting units are provided to cope with any such event which includes extensive damage to the wired lighting distribution

systems. An 8 hour battery capacity as specified in Section 8.5.3 for these lighting units will provide adequate capability to cope with such events.

Section 8.3.5 of Chapter 11 originally required that the circuits from the normal lighting system to the individual lighting fixtures be staggered as much as possible to ensure some lighting is retained in a room in the event of a circuit failure. The staff informed EPRI that, consistent with the staff position in its DSER on the evolutionary ALWR, the passive Requirements Document should also specify that, as a minimum, the staggered circuits to the normal lighting fixtures in safety-related areas and their access routes will be supplied from redundant electrical divisions. The passive Requirements Document should also provide design criteria that demonstrate that reasonable measures have been taken to address the survivability and the loss of lighting in those safety-related areas and access routes due to the effects of design-basis events. In its January 10, 1992, letter, EPRI provided the following response:

> In response to this question and the staff's position presented in Sections 8.2.1 and 8.2.2 of the draft safety evaluation report on Chapter 11 of the evolutionary plant, Section 8.3.5 of Chapter 11 for Volumes II and III will be revised as follows:
>
> "The circuits to the individual lighting fixtures shall be staggered as much as possible, with the staggered circuits fed from separate electrical divisions, so as to ensure that some lighting is retained in a room in the event of a circuit failure."
>
> Section 8.5 of Chapter 11 provides the design criteria that demonstrate that reasonable measures have been taken to address the loss of lighting due to the effects of design basis events. To further emphasize the intent of the design, the following requirements will be added to Section 8.5:
>
> "The emergency lighting system of the main control room shall be integrated with the normal lighting system and shall be designed so that alternate emergency lighting fixtures are fed from separate safety divisions."
>
> "The emergency lighting installations inside and outside the main control room shall be designed to remain functional during design basis events and in particular to withstand the seismic loads of a design basis earthquake."

Section 8.5.2 of Chapter 11 requires that the emergency lighting be accomplished by the following systems:

- main control room: emergency lighting system fed from safety uninterruptible (ac/dc) power supply
- outside main control room: dc self-contained, battery-operated lighting units

Section 8.5.2 also requires that the emergency lighting systems be operational during all emergency conditions, including SBO. The requirement for the emergency lighting in the main control room could be interpreted to mean that one uninterruptible power supply would be sufficient to power that lighting. However, in the foregoing EPRI response, EPRI committed to revise Section 8.5 to require that alternate emergency lighting fixtures in the main control room be fed from separate safety divisions. That change will resolve this issue.

The qualification of the emergency lighting systems both inside and outside the main control room also is not clear from the Section 8.5.2 requirement. Again, however, in its response above, EPRI has identified a change it committed to make in Section 8.5 that makes it clear that these systems are required to be designed to remain functional during design-basis events and design-basis earthquakes. That change will resolve this issue.

Lastly, the passive Requirements Document did not specifically specify how long the emergency lighting in the main control room must remain available following a loss of normal lighting. The staff informed EPRI that, consistent with the passive safety systems requirement that the passive systems remain available for 72 hours without ac power following an event, the emergency lighting in the main control room should also remain available for 72 hours without ac power following an event. The passive Requirements Document should specifically require the 72-hour capability so that there is no confusion regarding this capability. EPRI provided the following response relative to this issue in its January 10, 1992, letter:

> With regard to the 72 hour capability, it is the intent of the ALWR design to maintain the control room habitable for 72 hours without ac power following an event. To emphasize this intent, the following requirement will be added to Section 8.5 of Chapter 11:
>
> "The safety batteries used to supply power to the main control room emergency lighting system shall be sized to maintain the lighting levels specified in requirement 8.5.1 for at least 72 hours following a design basis event which includes the loss of all ac power sources."

In its DSER for Chapter 11, the staff stated that it was clear from EPRI's responses to the above issues and the changes EPRI committed to make, that the emergency lighting systems in the main control room will have an endurance of at least 72 hours on loss of normal lighting and will remain functional during a design-basis event and design-basis earthquake, and that the normal lighting systems will utilize staggered lighting fixtures as much as possible with the staggered fixtures fed from separate non-safety electrical divisions. However, because the emergency lighting systems outside the main control room still only have an endurance of 8 hours, a loss of the normal lighting systems, such as that caused by an SBO event or design-basis earthquake, could result in the loss of all lighting outside the main control room after 8 hours. It was not clear to the staff how much activity outside the main control room might be required in a passive plant design. The staff was aware, however, that in existing plant designs, event reports had demonstrated that substantial operator activity outside the main control room is often required during the reported events. This concern is closely tied in with the concern related to the RTNSS issue, that uncertainties in the passive plant

and its safety systems may require enhanced capability of the non-safety systems. The staff stated that potential failure or unreliabilities of the passive systems could likewise require unforeseen operator action outside the main control room on safety or non-safety equipment. In the DSER for Chapter 11, the staff stated that it would, therefore, address this issue again, following final resolution of the RTNSS issue discussed in Section 2.2.1 of this chapter. It also would confirm that EPRI's proposed changes were acceptably incorporated into the passive Requirements Document.

Revisions 2 and 3 to the passive Requirements Document acceptably incorporated EPRI's proposed changes given above. Therefore, this confirmatory issue is closed.

With regard to the issue involving the loss of plant lighting outside the main control room after 8 hours, the staff considers this to be an excellent candidate to be addressed under the process developed for resolving of the RTNSS issue the staff discusses in Section 2.2.1 of this chapter. As indicated in previous sections of this report, the staff will require that the non-safety standby power sources and ac distribution systems be addressed during the RTNSS process described in SECY-94-084. As a part of that process, the non-safety standby power sources must be assumed to be unavailable in the design-specific focused PRA. The staff will, therefore, ensure that the plant lighting outside the main control room is assumed available only from the emergency lighting system for 8 hours following the loss of the standby power sources and the ac distribution systems. If, under these assumptions, the focused PRA demonstrates that the design cannot meet the NRC safety goals and requirements because of the lack of lighting, the staff will pursue this issue further. This issue will, therefore, be addressed by the staff during its review of an individual application for FDA/DC.

8.2.2 Lighting Intensity of the Emergency Lighting System

Section 8.5.1 of Chapter 11 originally required that the emergency lighting system provide illumination units of at least 10 foot-candles in those areas of the plant where emergency operations will be performed that could require the reading of printed or written material or the reading of scales and legends. It states that these areas are typically control rooms or local control stations. In other areas of the plant, the emergency lighting is required to achieve a minimum illumination level of 2 foot-candles. The wording in the first part of this requirement was confusing and could be interpreted to mean that illumination units (such as battery-operated lighting units) would be provided that could put out 10 foot-candles of illumination at their source. The staff, therefore, asked EPRI that the wording be clarified to specify that the lighting provided must achieve a minimum illumination of 10 foot-candles on the printed or written material and the scales and legends. Also, control rooms and local control stations are not the only locations in the plant where emergency operations are performed that could require the reading of printed or written material or the reading of scales and legends. There are numerous locations throughout the plant where devices such as valves and switchgear have identification tags, warning labels, instruction labels, etc., that would require reading before their operation during an emergency event. The staff asked EPRI to make clear that these also must have a minimum

illumination of 10 foot-candles on the printed or written material and the scales and legends. EPRI provided the following response in its January 10, 1992, letter:

> This requirement was not intended to specify a source strength for the emergency lighting units, since the important attribute is the lighting (illumination) of the surfaces. Note that in the guidelines for control room design reviews, NUREG-0700, Section 6.1.5.4.c, the emergency lighting levels are specified simply ". . . to provide a minimum illumination level of 10 foot-candles at all work stations . . . ." The question may have resulted from the wording of the first sentence of the requirement. Accordingly, the first sentence of Section 8.5.1 will be revised as follows:
>
> "The emergency lighting system shall provide a minimum illumination level of 10 foot-candles at all work stations in the plant where emergency operations are performed which could require reading of printed or written material or the reading of scales and legends."
>
> It should also be noted that the first sentence of Section 8.5.1 indicates that the requirement to provide illumination levels of at least 10 foot-candles is not limited to the control rooms or local control stations but applies, in general, to the areas of the plant where emergency operations are performed which require reading of printed or written material.

In the DSER for Chapter 11, the staff stated that with regard to the lighting intensity, the revision proposed by EPRI was acceptable since it was now clear that the requirement was specifying an illumination level for the lighted area rather than the source strength for the emergency lighting units. With regard to the areas to which the requirement applies, the revision was acceptable with the understanding that the requirement was not limited to the control rooms or local control stations but applied, in general, to the areas of the plant where emergency operations are to be performed that require the reading of printed or written material. The insertion of this revision into the passive Requirements Document was identified as a confirmatory issue.

The change was made in Revision 2; therefore, this confirmatory issue is closed.

8.2.3 Lighting Quality of the Emergency Lighting System

In addition to the lighting illumination level specified in Section 8.5.1 of Chapter 11, the quality of the lighting is important in order to be able to decipher printed or written material and scales and legends. Although a label may have 10 foot-candles of illumination on it, it may be impossible to read if there is intolerable glare from the light source itself or if veiling reflections are produced off the label. This is a particular problem with directional or point light sources such as the incandescent lamps most often used in self-contained battery-operated lighting units such as those specified for the emergency lighting system. Although these problems can be minimized by initially locating and aiming the lights to achieve the best results, maintenance activities and relamping could end up redirecting the lights so that the foregoing problems are created or labels are in deep shadows without

the benefit of 10 foot-candles of illumination. Self-contained battery-operated lighting units are most often the best choice for providing lighting during and following fire-related events where the damage to the wired lighting distribution systems can be quite extensive. They are not, however, necessarily the best choice for events involving the loss of the normal lighting power source or those involving more localized damage to the normal lighting systems because of the above identified problems. The passive Requirements Document, however, specifies the self-contained battery-operated lighting units for all events involving the loss of the normal lighting systems, both fire related and non-fire related. The staff, therefore, asked EPRI to address the above concerns. EPRI provided the following response in its January 10, 1992, letter:

> As indicated in Section 2.3.2.9 of Volume III, Chapter 1, no operator action is required to maintain plant safety for at least 72 hours following a licensing basis event which results in a complete loss of all ac power sources. Certainly, no operator action outside the main control room will be necessary. Inside the main control room, the emergency lighting will be part of the normal lighting system and measures will be taken to prevent the type of problems described in this question (see Section 8.2.4).
>
> In the event that the control room must be evacuated, as a result of a fire for example, the normal sources of ac power and therefore normal lighting are expected to remain available at least in part. If extensive damage to the normal lighting distribution systems is assumed, however, the self-contained battery-operated lighting units specified in Chapter 11, Section 8.5.2, constitute the most appropriate alternative for providing emergency lighting.

As discussed in Section 8.2.1 of this chapter, it is not clear to the staff how much operator activity might be required outside the main control room during an event in a passive plant. On the basis of experience in existing plants, however, it is likely that at least some operator actions will be required outside the main control room. The staff also notes that, regarding the selection of remote or local control, the rationale portion of Section 3.4.4 of Chapter 10 of the passive Requirements Document states that within the main control rooms of current plants remote controls and displays are provided where local components would be completely adequate. It appears, therefore, that more controls may be located outside the main control room in advanced plants that might require the attention of an operator during a specific event. However, the staff is not sure what practical alternatives exist to the emergency lighting systems as proposed by EPRI in a passive plant design where the emergency lighting is provided from batteries. The staff, therefore, recommends that plant designers be attentive to the lighting quality of the emergency lighting systems, both outside and inside the main control room, by providing for sufficient optimally located lighting sources based on a careful study of the plant emergency lighting needs. Alternatives to the typical self-contained battery-operated lighting units should also be pursued by the plant designers.

8.2.4 Security Lighting System

Power supplies outside the protected area are vulnerable to intruders before detection at the protected area perimeter. The staff concludes that it is acceptable to require uninterruptible power for only those portions of the security lighting that are essential to plant protection following interruption of normal power.

Revision 2 to Section 8.4.2 of Chapter 11 also specifies a minimum illumination level in the isolation zone that exceeds the requirements of 10 CFR Part 73 (1 foot-candle rather than 0.2 foot-candle), and, therefore, is acceptable at this stage of the design. However, experience has shown that inattention to the integration of exterior lighting systems with the isolation zone's closed-circuit television (CCTV) system, particularly uniformity of lighting (e.g., light sources or reflections in the field of view, glare, blooming, and excessive light/dark ratios) can be detrimental to alarm assessment. NUREG/CR-1327, "Security Lighting Planning Document for Nuclear Fixed Site Facilities," recommends a light/dark ratio (the ratio of illumination level between the brightest area in the scene and the darkest area) of less than 6 to 1 for outdoor CCTV lighting. In its letter dated September 15, 1989, EPRI stated that the light/dark ratio is an engineering detail beyond the scope of the passive Requirements Document. The staff finds this position acceptable. It will determine if provisions for the integration of the exterior lighting systems with the CCTV system are acceptable during its review of an individual application for FDA/DC.

9 ELECTRICAL PROTECTIVE SYSTEMS

9.1 Functions and Description

Section 9 of Chapter 11 of the passive Requirements Document provides the system design and performance requirements for the station grounding systems, surge protection systems, cathodic protection systems, and heat tracing systems.

The grounding systems will be designed to protect personnel and equipment under normal and abnormal conditions. The major functions of the systems are summarized as follows:

- to protect personnel by eliminating or reducing shock hazards
- to protect equipment by minimizing transient overvoltages
- to provide low impedance path to ground for ground fault currents, lightning discharges, and switching surge currents and to facilitate protective relaying for fast clearing of ground faults

The grounding systems will include the following:

- ground mats which provide low-resistance interface with the earth
- plant electrical distribution system grounding equipment which is used to connect the electrical system's neutrals to ground
- equipment and structure grounding which is used to connect structures and equipment enclosures to ground
- instrumentation and control grounding equipment

The surge protection systems will be designed to protect plant equipment from exposure to overvoltage transients resulting from lightning strikes and switching operations. The surge protection systems will include surge arresters and capacitors and lightning protection equipment.

The cathodic protection system will control corrosion of underground and submerged metallic surfaces. It will be used to protect buried pipes, tanks, and other metallic equipment in contact with potentially corrosive sprays, water, and dissimilar metals against long-term degradation in order to avoid or reduce repair or replacement costs and plant shutdown.

The electrical heat tracing system will be designed to provide effective heating of fluids required for normal and transient plant operation. It will be applied to plant fluid systems, including piping, pumps, strainers, valves, and tanks, and will comprise electric heating cables, temperature controllers, power supplies, alarm and monitoring devices, and associated hardware.

9.2 Evaluation

The requirements in Section 9 of Chapter 11 of the passive Requirements Document are consistent with regulatory requirements and are, therefore, acceptable.

10 CONCLUSION

The staff concludes that the requirements established in Chapter 11 of the passive Requirements Document for the design of electric power systems do not conflict with current regulatory guidelines and are acceptable. However, by themselves, they do not provide sufficient information for the NRC staff to determine if the plant-specific design, operation, and arrangement of the electric power systems will be adequate. Applicants referencing the passive Requirements Document will be required to demonstrate compliance with the additional guidance in the SRP, or provide justification for alternative means of implementing the associated regulatory requirements.

Therefore, the staff concludes that Chapter 11 specifies requirements that, if properly translated into a design in accordance with the NRC regulations in force at the time the design is submitted, should result in a nuclear power plant whose electric power systems will perform as designed and have all the attributes required by the regulations to ensure that there is no undue risk to the health and safety of the public or to the environment.

APPENDIX A
DEFINITIONS AND ACRONYMS

Appendix A of Chapter 11 of the passive Requirements Document contains definitions of terms and acronyms. The staff has provided a consolidated list of acronyms in Volume 1 of this report.

CHAPTER 12, "RADIOACTIVE WASTE PROCESSING SYSTEMS"

1 INTRODUCTION

This chapter of the SER documents the NRC staff's review of Chapter 12, "Radioactive Waste Processing Systems," of the passive Requirements Document through Revision 5. Revision 5 was submitted to the NRC by letter dated October 1, 1993. Chapter 12 was prepared, under the project direction of EPRI and the ALWR Utility Steering Committee, by Bechtel Power Corporation; Duke Power Company; General Electric Company; Grove Engineering, Inc.; J. Vance and Associates; MPR Associates, Inc.; S. Levy Incorporated; Sargent and Lundy Engineers; Stone and Webster Engineering Corporation; Westinghouse Electric Corporation; and EPRI.

1.1 Review Criteria

Section 1 of Volume 1 of this report describes the review criteria used by the staff during its review of Chapter 12 of the passive Requirements Document.

1.2 Scope and Structure of Chapter 12

Chapter 12 of the passive Requirements Document defines the ALWR Utility Steering Committee's requirements for the design of the facilities used to process radioactive waste at ALWRs at which passive safety systems will be used.

The key topics addressed in the Chapter 12 review include gaseous radioactive waste processing systems, liquid radioactive waste processing systems, solid radioactive waste processing systems, and their associated support systems.

1.3 Policy Issues

During its review of Chapter 12 of the passive Requirements Document, the staff did not identify issues that involve policy questions for the technical areas discussed in this chapter, other than those already identified in Commission papers listed in Appendix B to Chapter 1 of this report.

1.4 Outstanding Issues

The DSER for Chapter 12 of the passive Requirements Document contained the following open issues:

Open Issues

P.12.0-1 source term basis for designing radioactive waste processing systems and evaluating offsite effluent radioactive nuclide concentrations (2.2.2)

P.12.0-2 basis for 2-minute delay requirement for BWR turbine gland seal system exhaust (3.3.1)

P.12.0-3 production sources for "essentially nonradioactive steam" (3.3.1)

P.12.0-4 discrepancy between Figure 12.3-1 and requirements in Chapter 13 (3.3.1)

P.12.0-5 use of post-filter downstream of charcoal adsorber in ventilation exhaust systems (3.3.3)

P.12.0-6 guidance regarding direct piping from radioactive plant systems to sumps or waste collection tanks (BWR) (4.2)

P.12.0-7 requirements for liquid radioactive waste processing system (LRWPS) filter housing and components (4.2)

P.12.0-8 requirements for LRWPS filters (4.2)

P.12.0-9 requirements for LRWPS ion exchangers (4.2)

Confirmatory Issues

None

The final disposition of each of these issues is discussed in detail in the appropriate section of this chapter, as indicated by the parenthetical notation following each issue. All outstanding issues identified in the DSER for Chapter 12 have been resolved.

1.5 Vendor- or Utility-Specific Items

The DSER for Chapter 12 contained the following vendor- or utility-specific items. the final disposition of each item is discussed in the appropriate section of this chapter, as indicated by the parenthetical notation following each item.

P.12.V-1 requirements for radioactive waste processing systems and effluent paths (2.2.8)

P.12.V-2 gaseous radioactive waste processing system hydrogen control design (3.3.4)

P.12.V-3 design of wet and dry waste shipping containers (5.5)

The following vendor- or utility-specific items remain and will be addressed during the staff's review of an individual application for FDA/DC. The staff's position on each item is provided in the appropriate section of this chapter, as indicated by the parenthetical notation following each item.

P.12.V-1 plant-specific designs for radioactive processing systems (2.2.1 and 2.2.8)

P.12.V-2 personnel radioactive exposure (2.2.4)

P.12.V-3 transfer of gaseous radwaste to plant vent (3.3.2)

P.12.V-4 compliance with SRP Section 11.3 for hydrogen control (3.3.4)

P.12.V-5 procedures for packaging and shipping processed solid wastes (5.5)

2 POLICY STATEMENTS AND KEY REQUIREMENTS

In Chapter 12 of the passive Requirements Document, EPRI specifies requirements for the radioactive waste processing systems. In general, Chapter 12 specifies that gaseous and liquid radioactive waste processing systems will connect to and receive radioactive fluids from those plant systems and components that may contain radioactive gases and liquids and require their removal for processing or disposal. These gaseous and liquid radioactive waste processing systems will terminate at recycle points, environmental discharge points, or solid radioactive waste system interfaces.

Although plant heating, ventilating, and air conditioning (HVAC) systems collect, filter, and discharge radioactive effluents, EPRI does not consider them to be gaseous radioactive waste processing systems. The EPRI design requirements for HVAC systems are discussed in Chapter 9 of the passive Requirements Document.

In general, Chapter 12 specifies that solid radioactive waste processing systems will begin at the discharge of potentially radioactive solids from processing equipment (e.g., cartridge filter vessels, charcoal adsorbers, high-efficiency particulate air filters), at points of collection of dry radioactive wastes resulting from operation and maintenance activities, and at points of discharge of radioactive resin slurries and sludges from the liquid radioactive waste processing system. The solid radioactive waste processing systems will terminate at the location where the solid radioactive wastes will be shipped off site.

2.1 Policy Statements

Section 1.5 of Chapter 12 of the passive Requirements Document contains those policies established by the ALWR Utility Steering Committee to be considered during the design and development of gaseous, liquid, and solid radioactive waste processing systems.

In Section 2.2.2 of the DSER for Chapter 12, the staff identified concerns related to EPRI's previously identified source term bases for the design of the radioactive waste processing system (RWPS) and evaluation of conformance of radionuclide concentrations in effluents to applicable 10 CFR Part 20 concentration limits. In letters dated January 13 and April 5, 1993, EPRI revised Sections 1.5.1 and 1.5.2 of Chapter 12 of the passive Requirements Document in response to the staff's concerns. EPRI also revised Sections 1.5.4, 2.2.2.1, and 3.2.1 of Chapter 12 to reflect consistency with revised Sections 1.5.1 and 1.5.2 and the deletion of Table 12.1-1 referred to in Revision 0 of Sections 1.5.4, 2.2.2.1, and 3.2.1.

Revised Section 1.5.1 of Chapter 12 states that three different bases have been selected for radionuclide source terms. These are to be used for the following:

(1) RWPS design; evaluation of annual average offsite radionuclide concentrations in liquid and gaseous effluents to demonstrate compliance with applicable 10 CFR Part 20 limits, and analysis of gaseous RWPS failure to demonstrate the system's compliance with NRC Branch Technical Position

ETSB 11-5, "Postulated Radioactive Release due to a Waste Gas System Leak or Failure." EPRI requires that the licensing source term for the above be based on a failed fuel rate consistent with SRP Sections 11.2, "Liquid Waste Management Systems," and 11.3, "Gaseous Waste Management Systems."

(2) Evaluation of expected offsite annual gaseous and liquid effluents during normal plant operation, including anticipated operational occurrences, and resulting gaseous and liquid pathway doses to the maximally exposed offsite individual to demonstrate compliance with 10 CFR Part 50, Appendix I dose limits. EPRI requires that the licensing source term for the above be based on American National Standards Institute American Nuclear Society (ANSI/ANS) 18.1. EPRI notes that the above document is periodically updated to reflect cumulative experience and that it may be revised before any design is finalized. EPRI requires that the version of ANSI/ANS 18.1 used for the above evaluation be checked for consistency with NUREG-0016 "Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents for Boiling Water Reactors (BWR-GALE Code)," and NUREG-0017 "Calculations of Release of Radioactive Materials in Gaseous and Liquid Effluents for Pressurized Water Reactors (PWR-GALE Code)," and, if there are any substantive differences, made consistent with the NRC evaluation.

(3) Evaluation of the RWPS performance for comparison with good neighbor policy goals. EPRI has given the source term basis for demonstrating compliance with the good neighbor policy goals in Section 1.5.2 of Chapter 12 of the passive Requirements Document.

Section 1.5.2 of Chapter 12 now defines the good neighbor policy as one that requires the plant to be a good neighbor to its surrounding environment and population by limiting gaseous and liquid radioactive releases from the plant to the environment during normal operation of the plant and by limiting offsite shipment of solid radioactive waste. EPRI requires that the plant be designed, constructed, and operated in such a manner that the amounts of radioactive gaseous and liquid wastes released to the environment and solid radioactive wastes shipped from the plant compare favorably with corresponding releases and shipments from the top 10 percent (in terms of low radioactivity releases) of plants of the same type (i.e., BWR or PWR) currently operating in the United States. EPRI requires the plant designer to use a source term basis of 0.025 percent failed fuel for PWRs and a noble gas total release rate of 15,000 μci/sec at 30 minutes' decay from the off-gas system for BWRs to demonstrate the plant's compliance with the best 10 percent criterion mentioned above.

Section 1.5.3 of Chapter 12 describes attributes for the baseline design and design options for liquid radioactive waste processing systems (LRWPSs) for the ALWRs. It states that the design basis for the LRWPS is to permit releases of radioactive liquids well below regulatory restrictions under expected conditions. The optional design is identified as one that accommodates "zero liquid release" and is to be used for sites where no radioactive liquid release to the environment from the plant will be allowed. The optional design at such sites may require additional processing equipment, both for the liquid and solid radioactive wastes, such as waste evaporators and associated waste solidification equipment.

Section 1.5.4 of Chapter 12 describes those features that would allow the radioactive waste processing systems to meet the objectives and goals of EPRI's ALWR program. These features include requirements proposed by EPRI for fuel performance, delay times for release of gaseous effluents, use of filtration and ion exchange technology in LRWPSs, reduction of irregular and unplanned leakage inputs through design considerations, consideration of the need to maintain a balance of the radioactive water within the plant, a need to maintain a plant tritium balance, and minimization of the generation of wet and dry solid wastes and the processing techniques that reduce the volumes of such wastes.

Section 1.5.5 of Chapter 12 lists the features that EPRI has determined will contribute to the reduction of radiation exposure to personnel working in the radioactive waste processing areas so that it will be as low as reasonably achievable (ALARA) and will meet the EPRI-stated goal of 100 person-rem/year for the plant. These features include the design for remote centralized operation, the selection and design of reliable and easily serviced equipment to minimize maintenance operations, the use of remote handling equipment, and consideration of the processing system configuration and radiation shielding locations to maintain personnel exposures ALARA.

Section 1.5.6 of Chapter 12 states that simplification will be achieved by the use of fewer pieces of equipment and less complicated equipment and controls. Charcoal adsorbers will be used at room temperature, and humidity will be controlled by cooling with the plant chilled water system. Gas decay tanks will be eliminated from the designs. Although EPRI concludes that evaporators should not be necessary and that processing should be performed through filtration and ion exchange, it has provided requirements for evaporators to be used on a case-by-case basis. EPRI also states that no solidification equipment is expected to be required and incinerators will not be included as a means for achieving the EPRI solid waste reduction goal.

Section 1.5.7 of Chapter 12 discusses the differences between BWRs and PWRs that affect radioactive waste processing system design.

Section 1.5.8 of Chapter 12 states that radioactive waste processing systems will be designed with sufficient process flexibility so that mobile equipment can be used and alternative internal processing paths will exist to accommodate future modifications.

Section 1.5.9 of Chapter 12 states that there is no expected need for long-term onsite storage of low-level radioactive waste. This section specifies that a minimum of 6 months of onsite storage space will be provided to accommodate unforeseen events and interruptions to the shipping of solid waste. A portion of the space will be shielded to allow radioactive decay of the more radioactive wastes for up to 6 months to enable shipping under less stringent conditions.

Section 1.5.10 of Chapter 12 states that the gaseous, liquid, and solid radioactive waste processing systems will not be classified as safety systems. Therefore, EPRI requires design features to ensure that highly radioactive fluids are not inadvertently transported to the radioactive waste processing systems under accident conditions.

2.2 Functions and Key Performance Requirements

Section 1.3 of Chapter 12 states that the overall functions of the radioactive waste processing systems are as follows:

- Separately collect and segregate in sumps, tanks, or bins (or drums), as appropriate, gaseous, liquid, and solid radioactive or potentially radioactive wastes through connections to the various plant systems and to drains serving radiation areas.
- Provide storage capability in order to accommodate process delays and disposal of processed wastes.
- Process the radioactive wastes in order to accumulate, remove, separate, reduce, or concentrate the radioactivity of the waste-containing streams or solids. The objective is to permit discharge, disposal, or recycling of the carrier stream and/or to permit packaging and offsite disposal of the resultant solid wastes.
- Safely and effectively process the wastes for safe, monitored discharge or disposal of effluents or solids from the station in compliance with the State and Federal regulations regarding discharges, transport, and disposal and to meet the ALWR goals.
- Permit adequate sampling and/or in-line measurement of input, process, and discharge streams for input into a data processing system to provide the operator with information for controlling and making decisions regarding routing, processing, disposal, and unusual occurrences. The processed data will also be used for reporting purposes.
- Serve as the principal means of controlling the tritium concentration in the reactor coolant in PWR plants.
- Ensure high-integrity boundaries for radioactive fluids to preclude their uncontrolled release to the environment.

2.2.1 Goals of Radioactive Release and Waste Reduction

Section 2.2.1 of Chapter 12 states that the radioactive waste processing systems will enable ALWR plant designs, which reference the passive Requirements Document, to meet the goals of reducing radioactive releases and reducing the volume of solid low-level waste from ALWRs. This section tasks the plant designer with the responsibility of demonstrating that the expected inputs to the radioactive waste processing systems and processing methods will result in outputs that meet the objectives and goals of the passive Requirements Document. The staff will ensure that the plant-specific designs of the radioactive waste processing systems comply with Commission regulations and regulatory guidance during its review of individual applications for FDA/DC.

Sections 11.2 and 11.3 of the SRP provide specific acceptance criteria regarding the inclusion in the radioactive waste treatment systems of all reasonably demonstrated technologies that can reduce radiation doses to the population for a favorable cost-benefit ratio. Compliance with these criteria

has not been identified as a requirement in Chapter 12 of the passive Requirements Document. However, Chapter 1, Appendix B (Table B.1-2) of the passive Requirements Document requires compliance with SRP Sections 11.2 and 11.3 and Regulatory Guide (RG) 1.110, Revision 0, "Cost-Benefit Analysis for Radwaste Systems for Light-Water Cooled Nuclear Power Reactors." This appendix addresses the use of available technology to reduce population radiation doses and provides guidelines for performing a cost-benefit analysis for radioactive waste processing systems for LWRs. Therefore, the staff concludes that this approach, consistent with the SRP, is acceptable. However, detailed radioactive waste processing system design features will be developed by the plant designer. The staff will review these design features to ensure compliance with Commission regulations and appropriate regulatory guidance during its review of individual applications for FDA/DC.

2.2.2 Source and Input Terms

Fuel source terms for the design and evaluation of radioactive waste processing systems are identified in Section 2.2.2 of Chapter 12. These fuel source terms provide the basis for system design and the evaluation of expected operation and performance.

Section 2.2.2.1 of Chapter 12 states that activation product source terms will be consistent with those given in NUREG-0016 for BWRs and in NUREG-0017 for PWRs.

Section 2.2.2.2 of Chapter 12 states that volumes and rates of release of gaseous and liquid radioactive wastes at the various source locations will be established on the basis of equipment performance characteristics; ANS/ANSI 55.1, 55.4, and 55.6; and operating experience.

Section 2.2.2.3 of Chapter 12 states that volumes of wet solid radioactive wastes will be derived from the performance parameters of the processes that produce them. Volumes of dry active solid radioactive wastes will be based on applicable operating experience.

In the DSER for Chapter 12, the staff concluded that EPRI's source terms for the design of the radioactive waste processing systems given in passive Requirements Document Table 12.1-1 were inconsistent with EPRI's stated compliance (passive Requirements Document, Chapter 1, Appendix B, Table B.1-2) with SRP Sections 11.2 and 11.3. Therefore, the staff identified the source term bases as an open issue. By letters dated January 13 and April 5, 1993, EPRI revised Sections 1.5.1, 1.5.2, 1.5.4, 2.2.2.1, and 3.2.1 of Chapter 12 and deleted Table 12.1-1 to resolve the DSER open issue. The staff concludes that the amended source term bases, as described in Section 2.1 of this chapter, are consistent with SRP Sections 11.2 and 11.3 and are, therefore, acceptable. Therefore, this DSER open issue is closed.

2.2.3 Releases of Radioactive Materials

In Section 1.5.4 of Chapter 12, EPRI states that operation under the licensing evaluation conditions specified will result in gaseous and liquid radioactive effluent releases that are within the limits specified in 10 CFR Part 50, Appendix I, for normal conditions and within the limits specified in 10 CFR Part 20 for normal and accident conditions.

Section 2.2.3 of Chapter 12 requires that the effluents normally released to unrestricted areas meet the goals in Chapter 12, Section 1.5.2 of the passive Requirements Document.

2.2.4 Personnel Radiation Exposure

Section 2.2.4 of Chapter 12 specifies that the radioactive waste processing systems for the ALWR will be designed so that personnel radiation exposure from operation and maintenance of these systems is consistent with the station requirement of 100 person-rem/year. Estimates of personnel radiation exposures to operate and maintain the radioactive waste processing systems are plant specific, and EPRI requires the plant designers to provide an estimate of expected personnel radiation exposures. The staff will review the vendor's or applicant's estimate of personnel radiation exposure during its review of an individual application for FDA/DC.

Section 2.2.7 of Chapter 12 requires each unit at a site to have its own radioactive waste processing systems, which will be operated remotely from a central radioactive waste control room. Section 2.2.10 of Chapter 12 requires that measuring elements, associated with radioactive waste processing system equipment having high radiation levels, be located outside the equipment cubicles in lower radiation areas, where practicable, to minimize personnel radiation exposures. These measuring elements will be static (with no moving parts) rather than dynamic devices, wherever possible, to reduce the potential for radioactive crud collection. These features are in accordance with the guidelines of RG 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable."

Section 2.2.11 of Chapter 12 requires the plant designer to perform a robotics evaluation to identify applications of robotics to inspection, surveillance, and maintenance of the radioactive waste processing systems. Inspection, surveillance, and maintenance activities associated with radioactive waste processing system operations have been significant contributors to overall plant personnel radiation exposures. In an attempt to reduce personnel radiation exposures, the plant designer will identify applications of robotics to these activities. In addition, the design features of the radioactive waste processing systems will facilitate the use of robotics. The use of robotics to reduce the radiation dose to plant personnel is in accordance with the guidance of RG 8.8.

2.2.5 Operating Conditions/Availability

Section 2.2.5 of Chapter 12 requires that the gaseous, liquid, and solid radioactive waste processing systems be available for operation during all phases of plant operation. However, for a BWR, the gaseous waste processing system is not required to be available for operation when the reactor is in the shutdown condition.

2.2.6 Operating Capacity

Section 2.2.6 of Chapter 12 requires that the radioactive waste processing systems have adequate capacity to handle wastes resulting from all plant operating and shutdown modes as well as anticipated operational occurrences.

It also requires that the radioactive waste processing systems accommodate infrequent, but unusual, conditions and large waste volumes.

As previously noted, Section 1.5.9 of Chapter 12 requires that an onsite storage facility be provided that can accommodate a 6-month storage period if solid radioactive waste is not shipped from the site.

2.2.7 Seismic Design and Quality Group Classification

Sections 2.2.8 and 2.2.9 of Chapter 12 require that radioactive waste processing systems and charcoal absorber supports conform to the Quality Group C and D requirements and the codes, standards, and seismic provisions of RG 1.143, Revision 1, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants." These sections also require that materials for pressure-retaining components conform to the guidelines of RG 1.143, Revision 1. In Section 4.6.3.3 of Chapter 6, EPRI requires that radioactive waste building design and equipment structural supports meet the guidelines of RG 1.143, Revision 1.

2.2.8 Control, Instrumentation, and Monitoring

Section 2.2.10 of Chapter 12 requires control and monitoring of the release of radioactive materials to the environment to be consistent with the requirements of the environmental monitoring system specified in Chapter 9 of the passive Requirements Document.

Additional process and effluent monitoring and sampling requirements are specified in several sections of Chapter 12 and in Chapters 3, 8, and 9 of the passive Requirements Document. However, the applicable portions of these chapters do not adequately specify all the requirements for process and effluent radiological monitoring instrumentation and sampling and do not adequately describe EPRI's commitment to comply with SRP Sections 11.5 and 11.5.A (EPRI committed to comply with these SRP sections in Chapter 1, Appendix B, Table B.1-2).

The staff concludes that, in general, the information in the applicable chapters lacks the minimum essential details regarding the required monitoring and sampling provisions for a number of waste streams. For example, it is not clear whether the ALWR design will include provisions for grab sampling and analysis of the following:

- effluent from the plant stack for noble gases, iodines, particulates, and tritium activity
- exhausts from auxiliary and radioactive waste buildings and fuel storage area for iodine activity
- effluents from liquid radioactive waste, service water, laundry and decontamination waste, secondary coolant waste, and turbine building drain waste systems for tritium activity

Also, some monitoring and sampling provisions identified in SRP Section 11.5, Tables 1 and 2, for the following waste streams are not included in the passive Requirements Document:

- automatic control features for terminating effluents from the steam generator blowdown system for a PWR and from the waste gas holdup system for an LWR

- continuous monitoring provisions for noble gas and iodine activity in the containment purge exhaust for a PWR

- continuous monitoring provisions for noble gas activity and grab sampling provisions for iodine activity in the process stream from the condenser evacuation system

- grab sampling provisions for iodine activity in the mechanical vacuum pump and turbine gland seal exhausts for an LWR and the containment purge exhausts for a PWR

- grab sampling provisions for iodine activity in the process streams from the evaporator, pressurizer, pretreatment liquid radioactive waste tank, and turbine building vent systems (filtering the discharges from these vent systems using high-efficiency particulate air filters and prefilters does not justify the omission of grab sampling provisions for iodine activity since these filters do not remove iodine in elemental and organic forms)

- continuous sampling provisions for effluent activity in the service water systems

The staff concludes that the passive Requirements Document does not adequately and explicitly address all associated regulatory requirements for control, monitoring, and sampling of radioactive waste processing systems and effluent paths. Therefore, the staff will review the plant-specific design for the control, monitoring, and sampling of radioactive waste processing systems and effluent paths against the criteria in SRP Section 11.5 during its review of an individual application for FDA/DC.

2.2.9 Fire Protection

Information pertaining to fire protection for radioactive waste processing systems in other parts of the passive Requirements Document is not repeated in Chapter 12. Fire protection for these systems is discussed in the following sections of the passive Requirements Document:

Solid Radioactive Waste Processing Systems

- Section 3.2 of Chapter 9 addresses the general requirements for fire protection (fire detection and suppression, fire hazards analysis, etc.) and requires conformance to SRP Section 9.5.1, "Fire Protection Program."

- Sections 8.2.1.1.4 (ventilation system penetrations), 8.2.1.1.17 (gas fire suppression), and 8.2.1.1.26 (charcoal filters) of Chapter 9 address the interface of fire protection systems with heating, ventilating and air conditioning systems.

- Section 5.4.2.8 of Chapter 12 requires fire detection and suppression for radioactive waste storage facilities.

Liquid Radioactive Waste Processing Systems

(System provisions for solid radioactive waste processing systems also apply.)

- Table 1.4.3, "Major Structural Design and Construction Codes," of Chapter 1 lists the national fire codes published by the National Fire Protection Association (NFPA). The National Electric Code, NFPA 70, addresses electrical equipment in hazardous environments such as off-gas recombiner rooms, where there is a potential for oil vapor or hydrogen leakage.

- Section 4.2.2.3.3 of Chapter 12 requires exclusion of oily waste to the extent practicable.

- Section 4.3.5.1 of Chapter 12 (mixed waste) prohibits the dry cleaning of protective clothing contaminated by radioactivity.

- Section 4.3.6 of Chapter 12 requires fire protection for stored mixed wastes that are flammable. (Note: The national fire codes define "combustible" and "flammable" as they apply to materials that will burn. The staff concludes that EPRI inadvertently used the word "flammable" rather than "combustible." In any case, the staff approves the provision of fire protection for combustible and flammable stored mixed wastes specified in Section 4.2.2.3.3 of Chapter 12.)

- Section 7.5.1.2 of Chapter 2 addresses the possibility of burning contaminated turbine lubricating oil in the plant auxiliary boiler.

Gaseous Radioactive Waste Processing Systems

(System provisions for solid and liquid radioactive waste processing systems also apply.)

- Section 3.2.3.2 of Chapter 12 addresses the requirements for maintaining waste gas streams free of flammable mixtures of hydrogen and oxygen. These requirements are necessary to protect the system from damage resulting from an explosion of the hydrogen-oxygen mixture and charcoal adsorbers from possible ignition.

- Section 3.3.7.2.5 of Chapter 12 addresses detection and suppression of fires in BWR charcoal beds located downstream of hydrogen recombiners in the off-gas system.

- Section 3.3.4 of Chapter 12 requires continuous ventilation in all areas where flammable gases, primarily hydrogen, might be present. The ventilation systems will provide a means of detecting and removing any flammable gases that might be present.

The staff concludes that the above provisions are consistent with the enhanced fire protection criteria discussed in Chapter 9 of this report.

2.3 Conclusion

The staff concludes that the requirements of Section 2 of Chapter 12 of the passive Requirements Document do not conflict with current regulatory guidelines and are acceptable. However, the requirements by themselves do not provide sufficient information to make a determination that the plant-specific design and arrangement will be adequate. Therefore, applicants referencing the passive Requirements Document will be required to demonstrate compliance with the additional guidance provided in the SRP, or provide justification for alternative means of implementing the associated regulatory requirements.

3 GASEOUS RADIOACTIVE WASTE PROCESSING SYSTEMS

3.1 Functions

Section 3.1 of Chapter 12 of the passive Requirements Document requires that a charcoal adsorber system be used for retaining fission gases for decay for both BWRs and PWRs. Section 3.1.2 requires that the BWR and PWR charcoal adsorber systems be designed to perform the following functions:

- receive and/or collect radioactive waste gases that originate in the reactor coolant system and that require processing by holdup for decay before they are released
- maintain the system nonflammable
- condition the gases to provide the moisture and temperature conditions necessary for the desired performance of the charcoal adsorbers
- retain the off gases for the desired time for the decay of fission products (xenon and krypton radioisotopes) resulting from fuel leakage and tramp uranium on fuel surfaces
- transport the remaining gases to a monitored heating, ventilating, and air conditioning (HVAC) vent for monitoring and release to the atmosphere
- provide against inadvertent release of significant quantities of gaseous and particulate radioactive material to the environment by providing non-safety-related monitoring and alarm functions
- ensure that in-plant occupational radiation exposures due to operation and maintenance of the off-gas systems are as low as is reasonably achievable

Section 3.1.2.2 of Chapter 12 states that the gaseous radioactive waste processing system for PWRs will not process the gases from the following systems, since they will only contain radioactive gas when there are concurrent fuel leaks and secondary steam generator tube leaks: main condenser air ejector, main condenser mechanical vacuum pump, turbine gland seal exhaust, and deaerator vent.

Section 3.1.2.3 of Chapter 12 requires that the BWR main condenser off-gas system be designed to perform the following functions:

- maintain hydrogen and oxygen mixtures noncombustible
- ensure that sufficient oxygen is present to react with the hydrogen added, if hydrogen water chemistry is used
- reduce the gas flow rate by the recombination of hydrogen and oxygen
- remove water vapor by condensation
- perform conditioning, decay, and transport functions

3.2 Performance Requirements

The following is a summary of the major performance requirements for the BWR and PWR gaseous radioactive waste processing systems (GRWPSs) and other BWR systems (turbine gland seal steam off-gas system and mechanical vacuum pump off-gas system) identified in Section 3.2 of Chapter 12 of the passive Requirements Document. Additionally, the following summary includes the major off-gas system and equipment requirements and the system control and instrumentation requirements for both BWRs and PWRs identified in Sections 3.3 and 3.4 of Chapter 12:

- Where the potential for an explosive mixture of hydrogen and oxygen exists, the system will be designed to maintain system integrity.
- For BWRs, the formation or buildup of explosive mixtures will be prevented, and the system will be designed to withstand the effects of a hydrogen detonation to permit operation to be resumed.
- For PWRs, parallel gas analyzers will be used to detect the formation or buildup of explosive mixtures, and the analyzers will annunciate locally and in the main control room to alert the operators to take remedial action.
- The off-gas process stream will be continuously maintained nonflammable, that is, by maintaining hydrogen levels below 4 percent by volume when oxygen is present (BWRs), or by maintaining oxygen levels below 4 percent by volume (PWRs).
- For BWRs, the hydrogen concentration will be maintained nonflammable by dilution with steam from the steam jet air ejector, and a dry air purge system will be provided upstream of each recombiner. The air purge flow rate and duration will be sufficient to purge any hydrogen from the system through the charcoal adsorbers.
- For PWRs, the gas in the system will be maintained slightly above atmospheric pressure.
- For PWRs, nitrogen will be available for inerting if the oxygen content of the gas stream exceeds 4 percent by volume.
- For BWRs, multiple charcoal adsorber beds will be arranged so that they can all be bypassed via a bypass bed.
- For PWRs, a minimum of two charcoal adsorber beds, in addition to a guard bed, will be provided with interconnections to bypass any one bed.
- Each BWR charcoal bed will be provided with a means for detecting and extinguishing a charcoal fire.
- Gases will be discharged to a monitored release point of the HVAC system.
- For BWRs, turbine seal off gases and vacuum pump off gases will be transported to and exhausted at a monitored plant vent.

- GRWPS discharges to the plant vent will be monitored by a GRWPS process monitor and provided with a high-level radiation alarm. Final releases will be monitored by the HVAC monitor at the plant vent.

The following features of the GRWPS are consistent with the guidelines specified in RG 8.8 for maintaining occupational radiation exposure as low as is reasonably achievable:

- Drain lines will be sized and continuously sloped to minimize the potential for plugging.

- Vents and drains will be designed to contain radioactive gases.

- Heat exchanger vent and drain valves will be located in a low-radiation area or they will be remotely operable.

3.3 Evaluation

3.3.1 BWR Turbine Seal Steam

BWR designs have typically relied on the use of nonradioactive steam for turbine gland seals to reduce the release of radioactive materials to the environment via the turbine gland seal exhausts. EPRI recognizes this objective and, therefore, states in Section 3.1.2.4 of Chapter 12 that essentially nonradioactive steam will be used to seal the BWR turbine gland seals. Further, in Section 7 of Chapter 2, EPRI has identified a requirement for providing an auxiliary steam system associated with an auxiliary steam boiler that, as one of its functions, will supply nonradioactive steam to turbine gland seals when main steam is unavailable.

In the DSER for Chapter 12, the staff identified some concerns relating to sealing steam for BWR turbine gland seals on the basis of its review of the original passive Requirements Document. Specifically, the staff was concerned that Figure 12.3-1 did not reflect the requirement stated in Chapter 13, Section 3.2.4.4, for automatic routing of the gland seal exhaust to the RWPS, should it be found to be highly radioactive. EPRI did not explain why a 2-minute delay line for the off gases from the gland seal system was identified as a requirement (Chapter 12, Section 3.2.4.1.2), even though the gland seal was to be sealed by essentially clean steam during power operation. EPRI did not explain why essentially clean steam was preferred to totally clean steam for sealing the turbine gland seals. EPRI did not explain how essentially clean steam would be produced. For the above reasons, the staff identified the above concerns as open issues.

In Revision 3 and in letters dated January 13 and April 5, 1993, EPRI amended Section 3.2.4.1.2 of Chapter 12, Section 3.2.4.4 of Chapter 13, and Figure 12.3-1. Amended Section 3.2.4.1.2 of Chapter 12, also requires a supply of essentially clean steam for sealing the turbine gland seals during normal plant operation. Additionally, it requires the capability to switch over to totally clean steam from an auxiliary boiler whenever gland seal exhaust is detected to be highly radioactive. Further, the requirement for a delay line has been deleted from the amended section; however, revised Figure 12.3-1 includes the provision of a delay pipe for the gland seal exhaust as an option for the plant designer should the designer see a need for the same for

decaying short-lived activation gases, such as nitrogen-16. Amended Section 3.2.4.4 of Chapter 13 requires the routing of the gland seal exhaust to a monitored release point, and the previous requirement to route it to the gaseous RWPS should the exhaust be detected to be highly radioactive has been deleted. The rationale for amended Section 3.2.4.1.2 states that since use of totally clean steam at all times would unnecessarily generate additional liquid radioactive waste because of the need to evaporate demineralized water, essentially clean steam is preferred to totally clean steam for sealing the turbine gland seals during normal plant operations. On the basis of the above, the staff finds that the amended sections and figure have satisfactorily addressed its concerns. Therefore, the proposed method to seal the BWR turbine gland seals is acceptable and the DSER open issue is resolved. However, since EPRI has not identified any specific method for producing essentially clean steam for sealing the turbine gland seals, the staff considers that EPRI is allowing the plant designer to have the flexibility to choose the method for producing essentially clean steam. Therefore, the staff will review applications for FDA/DC against 10 CFR Part 20 radionuclide concentration limits and 10 CFR Part 50 dose limits for offsite locations to ensure that the method chosen will produce essentially clean steam and consequently the gland seal exhaust will result in very small fractions of radionuclide concentrations and radiation dose limits for offsite locations.

3.3.2 Transfer of Gaseous Radioactive Wastes to Plant Vent

Regarding the transfer of gaseous radioactive wastes to a plant vent through appropriate HVAC systems, EPRI states in Section 3.2.5 of Chapter 12 that the passive ALWR designer will compile a list of vents from radioactive equipment throughout the plant that will enter the HVAC system. The list is to include the source (equipment or tank), the gas (normally air) flow rate, the radioactivity concentration, and the HVAC subsystem to which the vent is connected. The list also is required to include the sum of the air flows and radioactivities on a maximum and annual average basis. In addition, Section 3.4.3 requires the gaseous radioactive discharge from the GRWPS to be monitored by a GRWPS process monitor with a high-level radiation alarm. For BWRs, the staff expects that the list will include information not only on the radioactive discharges from the GRWPS but also on the discharges from other systems such as the turbine gland seal off-gas system and mechanical vacuum pump off-gas system. The staff will review individual applications for FDA/DC for compliance with appropriate regulations and regulatory guidance.

3.3.3 Installation of High-Efficiency Particulate Air Filters

In Section B.1.2.3 of Appendix B to Chapter 12 of the evolutionary Requirements Document and of the passive Requirements Document, EPRI explicitly recommends that high-efficiency particulate air (HEPA) exhaust filters be eliminated in GRWPS designs. In addition, Figure 12.3-1 of Chapter 12 of the passive Requirements Document depicts a BWR GRWPS without HEPA filters. Routinely, HEPA filters have been provided downstream of charcoal delay beds in the off-gas systems of BWRs to remove radioactive particulates (resulting from decay of noble gas radioisotopes when they are processed in the charcoal delay beds) from the processed off-gas effluent stream.

By letter dated December 20, 1991, EPRI addressed the staff's concern regarding the elimination of HEPA filters downstream of charcoal delay beds for the

evolutionary ALWRs. EPRI stated that the current practice was a holdover from past off-gas system designs that had only a 30-minute-delay pipe before the stack. Current off-gas system designs involve charcoal delay beds of several feet in diameter and flow rates in the system that are low (typically 10 to 40 standard cubic feet per minute). Therefore, the flow velocities will be too low (much less than 1 ft/sec) to transport significant amounts of radioactive particles of any size. An industry survey conducted for EPRI confirmed its rationale for eliminating the downstream HEPA filters in GRWPSs. Further, EPRI concluded that its rationale for eliminating the downstream HEPAs was equally valid for PWR GRWPSs that use charcoal holdup of noble gases. For the above reasons, with which the staff agrees, the staff concluded that elimination of the HEPA filter for evolutionary plants was acceptable. The staff concludes that EPRI's justification for eliminating the downstream HEPA filters in GRWPSs of evolutionary ALWRs is equally applicable for passive ALWRs and, therefore, the EPRI recommendation for the passive ALWRs is acceptable.

Because EPRI's position might have a bearing on gaseous radioactive waste management, the staff also reviewed EPRI's requirements concerning the treatment of gaseous effluents from plant ventilation exhaust systems. Compliance with 10 CFR Part 50, Appendix I offsite radiation dose limits for these effluents may require charcoal adsorbers and/or HEPA filters in the exhaust systems to reduce the release of radioiodine, in elemental and organic forms, and radioactive particulates to the environment. For a ventilation exhaust system that requires a filtration system, including both the charcoal adsorber and the HEPA filter, the staff allows the efficiencies for the removal of radioiodine and radioactive particulates from the effluent stream specified in RG 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," only if the filtration system is designed, tested, and maintained in accordance with the RG 1.140 guidelines, as stated in SRP Section 11.3. The guide, in turn, states that HEPA filters should be provided, one upstream of the charcoal adsorber and another downstream of the adsorber, with the downstream one for collecting carbon fines. The staff considers that it is incorrect to assume the regulatory guide efficiencies for the removal of elemental and organic iodine from a ventilation exhaust if there is no appropriate provision downstream of the charcoal adsorber for collecting potential carbon fines. The staff notes that EPRI discusses ventilation exhaust treatment systems as they relate to charcoal filtration in Section 8.2.1.1.26 of Chapter 9 of the passive Requirements Document. In addition to a pre-filter and a HEPA filter upstream of the charcoal adsorber, the original passive Requirements Document called for a post-filter (of unspecified efficiency) instead of a HEPA filter downstream of the charcoal adsorber to collect carbon fines. Such a post-filter was required only if the discharge was to other safety-related equipment or occupied spaces. In the DSER for Chapter 12, the staff concluded that EPRI had not adequately addressed the issue of collection of carbon fines downstream of charcoal adsorbers and identified this as an open issue.

In its letter dated January 26, 1993, EPRI revised Section 8.2.1.1.26 of Chapter 9 and its rationale to require that the filter system components be designed in accordance with ANSI/ASME N509-1989 and RG 1.140 standards. Specifically, it mandates the provision of a post-filter downstream of the charcoal adsorber, irrespective of the location of discharge. Further, the

amended rationale states explicitly that the efficiencies given in Section 5.3.1 of ANSI/ASME N509-1989 for pre-filter and post-filter are applicable. The staff notes that the subject section of the standard requires 95-percent efficiency for the post-filter downstream of the charcoal adsorber. The staff concludes that the revised section and its rationale ensure that the carbon fines will be collected in a satisfactory manner and are, therefore, acceptable. Therefore, this DSER open issue is closed.

3.3.4 Hydrogen Control

SRP Section 11.3, Acceptance Criterion II.B.6, provides the features required to protect the GRWPS from potentially explosive mixtures of hydrogen and oxygen. Section 3.2.3 of Chapter 12 of the passive Requirements Document specifies design criteria that are intended to prevent the formation of explosive hydrogen and air mixtures and to ensure that the GRWPS is designed to withstand the effects of detonations. EPRI states that the off-gas process stream will be continuously maintained nonflammable by limiting either the hydrogen volumetric concentration to less than 4 percent for BWRs when oxygen is present, or the oxygen volumetric concentration to less than 4 percent for PWRs. For BWRs, the oxygen concentration will be maintained nonflammable by dilution with steam following the second stage of the steam jet air ejector. In the case of hydrogen water chemistry for BWRs, when waste gases contain an excess of hydrogen, stoichiometric amounts of oxygen will be injected upstream of the recombiner. In PWRs, where the off gases are infrequently received from the reactor coolant and auxiliary building drain tanks and where the boron recycle steam is the only plant degassifier providing waste gas to the GRWPS, gas analyzers connected to the annunciators will be provided to warn the operators to take remedial actions.

Table B.1-2 of Appendix B to Chapter 1 of the passive Requirements Document shows that the ALWR design will comply with SRP Section 11.3. However, the requirements in Section 3.2.3 of Chapter 12 are not sufficient for the staff to conclude that the GRWPS will be designed to be consistent with Acceptance Criterion II.B.6 of SRP Section 11.3. For example, Section 3.2.3 of Chapter 12 does not state the criterion for minimum design pressure of the GRWPS for a BWR to withstand the effects of a hydrogen explosion, nor does it provide the needed number, type, and location of analyzers for monitoring hydrogen and oxygen. Table 12.3-1 of the passive Requirements Document does not provide sufficient details for the staff to conclude that the monitoring provisions will be consistent with the acceptance criterion of the SRP.

Although EPRI committed, in Chapter 1 of the passive Requirements Document, to comply with SRP Section 11.3, Chapter 12 of the passive Requirements Document does not address all the criteria in the SRP. The staff expects that applicants referencing the passive Requirements Document will comply with the SRP, as committed to in Chapter 1, and concludes that this is acceptable. The staff will review individual applications for FDA/DC against the SRP criteria discussed above.

3.4 Conclusion

The staff concludes that the requirements of Section 3 of Chapter 12 of the passive Requirements Document do not conflict with current regulatory guidelines and are acceptable. However, the requirements by themselves do not

provide sufficient information to make a determination that the plant-specific design and arrangement will be adequate. Therefore, applicants referencing the passive Requirements Document will be required to demonstrate compliance with the additional guidance provided in the SRP, or provide justification for alternative means of implementing the associated regulatory requirements.

4 LIQUID RADIOACTIVE WASTE PROCESSING SYSTEM

4.1 Functions

Section 4.1.2 of Chapter 12 of the passive Requirements Document requires that the liquid radioactive waste processing system (LRWPS) be designed to perform the following functions:

- classify and segregate wastes into subsystems for collection and efficient and economical processing
- provide the capacity to accumulate and process liquid radioactive wastes produced in the plant during normal operation and anticipated operational occurrences, including shutdown, refueling, and maintenance, without affecting plant availability
- provide alternative or redundant processing paths to ensure LRWPS and plant availability and continuous waste-receiving capability, including provisions for the use of mobile equipment or of cross connections
- provide for the transfer of filter backwashes, tank sludge, and spent resin to the solid radioactive waste processing system
- produce treated waste of acceptable quality for reuse within the plant, as appropriate, particularly that from BWR equipment drains
- produce treated effluents that can be discharged to the environment within the ALWR goals (for these goals, see Section 2.1 of this chapter of this report as it relates to Section 1.5.2 of Chapter 12)
- control and monitor LRWPS releases of chemical and radioactive materials to the environment in accordance with the applicable regulatory requirements
- protect plant personnel from radiation exposure and incorporate the basic ALARA objectives by, for example, the use of automated systems, appropriate arrangement, reliable and readily operable and maintainable equipment, shielding, and remotely operated instrumentation and controls.

4.2 LRWPS Requirements

The following is a summary of the major performance, arrangement, and equipment requirements for the LRWPS given in Sections 4.2, 4.4, and 4.5 of Chapter 12:

- Radioactive wastes will be segregated according to the following subsystems: equipment drains, floor drains, chemical wastes, detergent wastes, and mixed wastes.
- A single monitored pipe from the LRWPS sample tanks will be routed to the environment.

- A minimum flow of dilution water for radioactive waste discharges will always be available.

- All liquid radioactive waste will be discharged to the environment via tanks where the contents of the tanks will be mixed and sampled and proven acceptable before they are discharged.

- Each subsystem of the LRWPS will have sufficient process flow and storage capacity to process the inputs during normal operation and during anticipated operational occurrences.

- Provisions will be made for the collection of leakage and spills and for the control of radioactive gases and particles in locations where mobile equipment may be used.

- Tanks will be designed to prevent uncontrolled releases of radioactive material due to spillage.

- Hydrogen-containing wastes in PWRs will be collected in covered drain tanks or routed directly via the chemical and volume control systems to the borated waste processing subsystem.

- To eliminate a potential source of airborne radioactivity, liquid radioactive wastes from non-hydrogen-bearing (PWR) radioactive or potentially radioactive plant systems will be piped directly from radioactive plant systems to sumps or collection tanks.

- Equipment drain sumps in controlled access areas will be provided with curbs to prevent them from becoming contaminated in the event of flooding from adjacent floors.

- Floor drain piping will be routed and/or sealed to prevent cross flow of airborne radioactivity between building rooms and/or compartments where such cross flow is undesirable.

- Piping for concentrated chemical solutions will be sized to prevent plugging during resin transfer and will have provisions for flushing to keep it free of solids after use; the piping will be designed to preclude solidification of the solution in the lines and heat traced, as appropriate.

- Valves used in the LRWPS must not act as crud traps. Materials used for valve packing will have adequate radiation resistance to minimize the replacement of packing.

- Pumps will be flange-connected to facilitate their removal and replacement.

- Evaporators (when used) will be designed for ease of disassembly and assembly and replacement of internal components. Also, they will be capable of remote operation during maintenance activities.

- Remote or semi-remote handling techniques and equipment will be used to the extent practicable in the operation and maintenance of the system.

- All tanks will be provided with outside ladders and manholes sized to permit easy access for personnel wearing anticontamination clothing. To minimize the accumulation of crud on tank bottoms (and the ensuing increase in radiation levels), tank bottoms will be sloped or dished with a drain line at the lowest point of the tank and crevices and crud traps will be excluded during fabrication.

- Evaporators will be capable of being drained and flushed after use to prevent crud buildup and will be provided with connections for chemical decontamination. Radioactive components of the evaporator complex will be arranged in separate shielded areas to reduce radiation exposure from adjacent components during maintenance activities.

- Valves and instrumentation in lines servicing the evaporator will be located outside the evaporator enclosure so that maintenance on or operation of these components can be performed in a lower radiation field. In addition, all evaporator sampling points will be shielded to minimize personnel exposure.

In the DSER for Chapter 12, the staff concluded that EPRI had not provided adequate guidance or requirements regarding direct piping from radioactive plant systems to sumps or radioactive waste collection tanks to eliminate potential sources of airborne radioactivity for BWRs and identified this as an open issue. In letters dated August 19, 1992, and January 13, 1993, EPRI extended the direct piping design requirement to include BWRs. The staff concludes that this is acceptable; therefore, this open issue is closed.

In the DSER for Chapter 12, the staff concluded that EPRI had not provided adequate guidance or requirements for: (1) filter housing and components; (2) various types of filters in the LRWPS, including the capability to disassemble, reassemble, and replace internal components; and (3) design considerations and operational requirements for ion exchangers (e.g., addition and removal; retention; strainers; underdrains; and disassembly, assembly, and replacement of internal components). In DSER for Chapter 12, the staff identified each of these three deficiencies as open issues. In letters dated August 19, 1992, and January 13, 1993, EPRI responded to all three of these open issues. Section 4.5.9 of Chapter 12 has been revised to cross reference the requirements of Section 12.9 of Chapter 1, which deals with the design requirements for filter housing and components, various types of filters, and ion exchangers.

The staff has reviewed Section 12.9 of Chapter 1 and finds that it provides adequate guidance or design requirements for various types of filters, including filter housing (cartridge type, bag type, pressure precoat-vertical tube type), filter/demineralizer type (stacked-disc type and cross-flow type), and ion exchangers. Therefore, these three DSER open issues are closed.

4.3 Conclusion

The staff concludes that the requirements of Section 4 of Chapter 12 of the passive Requirements Document do not conflict with current regulatory guidelines and are acceptable. However, the requirements by themselves do not provide sufficient information to make a determination that the plant-specific design and arrangement will be adequate. Therefore, applicants referencing

the passive Requirements Document will be required to demonstrate compliance with the additional guidance provided in the SRP, or provide justification for alternative means for implementing the associated regulatory requirements.

5 SOLID RADIOACTIVE WASTE PROCESSING SYSTEM

5.1 Definition

The solid radioactive waste processing system will include the wet solid waste processing system, the dry solid waste processing system, and the onsite storage facility. According to the passive Requirements Document, wet solid wastes will consist of spent resins, filter backwash sludges, and tank sludges. Dry solid wastes will consist of compactible wastes, such as rags, paper, and clothing, and noncompactible wastes, such as contaminated tools, discarded radioactive equipment parts, maintenance wastes, and PWR filter cartridges.

5.2 Wet Solid Waste Processing System

5.2.1 Functions

Section 5.2.1 of Chapter 12 of the passive Requirements Document requires that the wet solid waste processing system be designed to perform the following functions:

- Transport, receive, store, and process sludges from filter backwash and tank settling, and resins from ion exchangers. Processing will include sludge concentration in phase separators, with routing of the decant to the liquid radioactive waste processing system.
- Transfer the spent resins and sludges and dewater them within a shipping container.
- Return removed fluids to the liquid radioactive waste processing system.
- Cap or otherwise seal the container.
- Sample the waste in a representative location to determine its classification in accordance with 10 CFR Part 61.
- Sample the external surface of containers (or shield for shielded containers) for surface contamination.
- Transport filled containers to the onsite storage facility if shipment is temporarily not possible.

5.2.2 System Requirements

The following is a summary of the major performance, system, and equipment requirements for the wet solid waste processing system described in Sections 5.2.2 and 5.2.3 of Chapter 12 of the passive Requirements Document:

- Spent ion exchange resins will be collected by type (bead or powder) and/or by expected radiation level in either spent resin or phase separator tanks.

- The waste sludge phase separator tank will collect filtered backwash sludges and tank sludges.

- The plant designer will establish the dewatering system capacity and process schedule to enable a tank of wet wastes to be processed and shipped within one-half the time interval between successive discharges into the spent resin tank or phase separator tank.

- The wet solid waste processing system will be capable of producing a product that satisfies the waste characteristics requirements of 10 CFR 61.56, pertinent State regulations, and disposal site requirements.

- The plant designer will define the set of process operating parameters and associated tolerances that ensure that the waste characteristics requirements can be met for each type of wet solid waste processed.

- The plant designer will define a test program that demonstrates that the waste characteristics requirements can be met for each type of wet solid waste processed. Testing is required for new process equipment or different waste forms. If the process equipment and waste types are identical to those at operating plants, the designer may substitute the results of operating experience for performance testing.

- The plant designer will design the dewatering equipment, as well as the permanent piping and valves, for receipt of spent resins and sludges and return of process effluents and will provide for the interface and performance requirements for the use of mobile equipment for the dewatering of resins and sludges, if desired.

- Means will be provided to permit future transfer of spent resins and sludges to an in-plant solidification facility.

- Piping used for the hydraulic transport of slurries, such as ion exchange resins, filter backwash sludges, and waste tank sludges, and waste concentrates (if present) will be sized to prevent plugging during transport and will have provisions for flushing.

- Means will be provided to permit a radiation survey of the contents of each waste container so that the waste can be classified in accordance with 10 CFR 61.55. Radiation surveys of container contents will be performed by using gamma scans from outside the container, if possible, rather than by monitoring waste removed from the container.

- Means will be provided to obtain samples periodically for determining and verifying the scaling factors used for waste classification.

- The design of the equipment for dewatering the radioactive slurry will incorporate features such as the capability to hook up and disengage the filling head to the container either remote automatically or remote manually to minimize operator exposure. Additionally, the design will permit the above operations to be performed manually. The slurry container will be designed to prevent slurry overflow and will be vented to minimize the spread of airborne contamination.

- The filled containers will be properly shielded, as needed; filled containers which are not shielded will be remotely transported to minimize operator radiation exposure.

The expected isotopic distribution and resulting radiation levels to be used for shielding evaluations of systems containing wet solid wastes are not within the scope of the passive Requirements Document and will be defined by the plant designer.

5.3 Dry Solid Waste Processing System

5.3.1 Functions

Section 5.3.1 of Chapter 12 of the passive Requirements Document requires that the dry solid waste processing system be designed to perform the following functions:

- segregate wastes at the source locations
- transport dry solid wastes to the radioactive waste facility
- provide interim storage for incoming dry solid waste
- sort dry solid waste (compactible from noncompactible, waste with activity too high for hand sorting, nonradioactive waste, and waste that will be reused)
- compact dry solid waste that is compactible
- package both compactible and noncompactible dry solid waste into storage or shipping containers
- transport filled containers to the shipping vehicle or the onsite storage facility
- monitor the waste to determine its classification in accordance with 10 CFR Part 61

5.3.2 System Requirements

The following is a summary of the major performance, system, and equipment requirements for the dry solid waste processing system given in Sections 5.3.2 and 5.3.3 of Chapter 12 of the passive Requirements Document:

- The dry solid waste processing system will be capable of producing a product that satisfies the waste characteristics requirements of 10 CFR 61.56, pertinent State regulations, and disposal site requirements.
- Means will be provided to permit the monitoring of dry solid waste containers to determine the classification of the waste in accordance with 10 CFR 61.55.

EPRI has provided the following requirements in accordance with the guidance of RG 8.8:

- Unsorted dry solid waste will be stored in a separate room with one area for high-activity waste and another for low-activity waste. Movable shielding will be provided to separate the high- and low-activity waste storage areas.

- Equipment needed for transporting filled waste shipping containers will be shielded to minimize operator exposure.

- Both the sorting table (the table for sorting dry solid wastes by hand) and the compactor for dry solid waste will have dedicated air filtration systems with high-efficiency particulate air filters to prevent the spread of airborne contamination and to limit respirator use by personnel, as appropriate.

5.4 Onsite Storage Facility

5.4.1 Functions

Section 5.4.1 of Chapter 12 of the passive Requirements Document requires that the onsite storage facility be designed to perform the following functions:

- Provide for at least 6 months of storage of all packaged wet and dry solid wastes.

- Provide shielding as required.

- Provide for transport, placement, and removal of packaged wastes within the premises.

- Provide means for collecting spilled liquids and releasing airborne radioactivity via monitored release paths.

- Provide shielded space for radioactive decay of packaged wastes before they are shipped. This is included in the 6-month space requirement.

5.4.2 Performance Requirements

The following is a summary of the major performance requirements for the onsite storage facility identified in Section 5.4.2 of Chapter 12 of the passive Requirements Document:

- High- and low-radiation waste containers will be stored in separate shielded storage areas within an onsite storage facility that will be in close proximity to the radioactive waste facility and within a common low-level, controlled access area. The storage areas will be shielded so that radiation exposure from the stored containers will not restrict access to onsite areas outside the storage area and will not exceed 2.5 mrem/hour for adjacent areas.

- The storage facility will have curbs to contain any spills of solids or liquids (e.g., sludges or dewatered resins) resulting from the inadvertent breach of a waste container, and the building atmosphere will be vented through the heating, ventilation, and air conditioning system to prevent the uncontrolled release of any airborne radioactivity.

- The high-radiation storage area will be provided with a remote viewing capability to verify handling operations and to permit identification of the containers.

- The required storage volume will be based on the maximum number of full shipping or storage containers used in 6 months and the number of containers required to fill one transport vehicle, plus a 25-percent margin.

- The storage area will be located indoors.

- The facility will have provisions for collecting any spilled radioactive liquids and returning them to the liquid radioactive waste processing system.

5.5 Shipping Containers

Chapter 12 of the passive Requirement Document, as amended by EPRI's letter dated January 13, 1993, contains identical interface requirements for the plant designer in Sections 5.2.1.4 and 5.3.1.4. The interface requirement calls for the plant designer to interface with the shipping container supplier and disposal site to ensure compatibility of the designs of the wet solid waste and dry solid waste systems with the loading, system connections, and waste form requirements of the shipping container supplier and the disposal site. The amended sections are an improvement over the original version of the passive Requirements Document because they clearly define the scope of the interface requirement and further extend its applicability to dry solid waste system design (the original version identified interface requirement only for the wet solid waste system design). The staff concludes that this interface requirement is acceptable. It will review the description of procedures for the packaging and shipment of processed solid wastes to an approved disposal site during its review of an individual application for FDA/DC or for a COL.

5.6 Conclusion

The staff concludes that the requirements of Section 5 of Chapter 12 of the passive Requirements Document do not conflict with current regulatory guidelines and are acceptable. However, the requirements by themselves do not provide sufficient information to make a determination that the plant-specific design and arrangement will be adequate. Therefore, applicants referencing the passive Requirements Document will be required to demonstrate compliance with the additional guidance provided in the SRP, or provide justification for alternative means for implementing the associated regulatory requirements.

6 CONCLUSION

The staff concludes that the requirements established in Chapter 12 of the passive Requirements Document for the design and arrangement of the radioactive waste processing systems do not conflict with current regulatory guidelines and are acceptable. However, by themselves, they do not provide sufficient information to make a determination that the plant-specific design and arrangement of the radioactive waste processing systems will be adequate. Therefore, applicants referencing the passive Requirements Document are required to demonstrate compliance with the additional guidance provided in the SRP, or provide justification for alternative means of implementing the associated regulatory requirements.

Therefore, the staff concludes that Chapter 12 specifies requirements that, if properly translated into a design in accordance with the NRC regulations in force at the time the design is submitted, should result in radioactive waste processing systems that will have all the attributes required by the regulations to ensure that there is no undue risk to the health and safety of the public or to the environment.

APPENDIX A

DEFINITIONS AND ACRONYMS

Appendix A of Chapter 12 of the passive Requirements Document contains definitions of acronyms. The staff has provided a consolidated list of acronyms in Volume 1 of this report.

APPENDIX B
SUMMARY OF DESIGN AND OPERATING TECHNIQUES TO REDUCE RADIOACTIVE WASTE

Appendix B to Chapter 12 of the passive Requirements Document provides a summary of design and operating techniques to reduce the generation of liquid, solid, and gaseous radioactive waste volumes; minimize liquid waste contamination; minimize area contamination; ease cleanup of contaminated areas; and provide equipment decontamination capability. Additionally, this appendix summarizes EPRI's perception of good management and operating practices for meeting the objectives mentioned above.

The appendix contains guidelines that may be useful to the plant designer. EPRI has included the appendix for information only and does not consider it as part of the Chapter 12 requirements (Chapter 12, Section 1.2, Page 12.1-3). Therefore, the staff considers all the design and operating techniques identified in the appendix as suggestions or recommendations to the plant designer, except for a few, which are identified as requirements in other parts of Chapter 12. The staff has no position on the recommendations that are not identified as requirements elsewhere in the passive Requirements Document.

CHAPTER 13, "MAIN TURBINE GENERATOR SYSTEMS"

1 INTRODUCTION

This chapter of the SER documents the NRC staff's review of Chapter 13, "Main Turbine-Generator Systems," of the passive Requirements Document through Revision 5. Revision 5 was submitted to the NRC by letter dated October 1, 1993. Chapter 13 was prepared, under the project direction of EPRI and the ALWR Utility Steering Committee, by Duke Power Company; General Electric Company; MPR Associates, Inc.; S. Levy Incorporated; Science Applications International Corporation; Westinghouse Electric Corporation; and EPRI Corporation.

1.1 Review Criteria

Section 1 of Volume 1 of this report describes the review criteria that the NRC staff used during its review of Chapter 13 of the passive Requirements Document.

1.2 Scope and Structure of Chapter 13

Chapter 13 of the passive Requirements Document provides basic design requirements for the main turbine, the main generator, and their respective support subsystems.

The key topics addressed in the Chapter 13 review include EPRI-proposed design requirements for

- availability
- planned outage schedule
- duty cycle
- abnormal operating conditions
- equipment environment.

1.3 Policy Issues

During its review of Chapter 13 of the passive Requirements Document, the NRC staff did not identify issues that involve policy questions for the technical areas discussed in this chapter, other than those already identified in the Commission papers listed in Appendix B to Chapter 1 of this report.

1.4 Outstanding Issues

The DSER for Chapter 13 of the passive Requirements Document contained the following outstanding issues:

Open Issues

P.13.0-1 effect of turbine missile generation probability on safety-related systems or components (3.1.4)

P.13.O-2 inservice inspection interval for turbine steam valves (3.3)

Confirmatory Issues

P.13.C-1 compliance with Regulatory Guide 1.29 seismic design classification (3.1.1)

P.13.C-2 compliance with Standard Review Plan quality group classifications (3.1.2)

The final disposition of each of these issues is discussed in detail in the appropriate section of this chapter, as indicated by the parenthetical notation following each issue. All outstanding issues identified in the DSER for Chapter 13 have been resolved.

1.5 Vendor- or Utility-Specific Items

The DSER for Chapter 13 contained the following vendor- or utility-specific items. The final disposition of each item is discussed in the appropriate section of this chapter, as indicated by the parenthetical notation following each item.

P.13.V-1 use of seismic experience database and analysis of safe shutdown earthquake loading conditions (3.1.1)

P.13.V-2 one-piece rotor design part-machining inspection requirements (3.1.5)

P.13.V-3 prototype testing of new or significantly changed turbine-generator designs (3.1.6 and 4.1.1)

The following vendor- or utility-specific items remain and will be addressed during the staff's review of an individual application for FDA/DC. The staff's position on each item is provided in the appropriate section of this chapter, as indicated by the parenthetical notation following each item.

P.13.V-1 inspection interval and maintenance program for turbine rotors (3.1.3)

P.13.V-2 adequacy of turbine control system (3.3)

P.13.V-3 recommendations for plant maintenance (3.3)

2 POLICY STATEMENTS AND KEY REQUIREMENTS

2.1 Policy Statements

Chapter 13 of the passive Requirements Document describes requirements for the turbine-generator systems. These requirements apply to ALWRs at which passive safety systems will be used. Section 1 of Chapter 13 contains those policies the ALWR Utility Steering Committee established that relate to the design approach to be used in developing the main turbine system and the main generator system. According to EPRI, the main turbine system will include the main turbine, the lube oil system, the turbine control system, the gland seal system, and their associated instrumentation. The main generator system will include the main generator, the excitation system, the stator cooling water system, the hydrogen cooling system, the hydrogen seal oil system, the hydrogen and carbon dioxide system, the generator control system, and their associated instrumentation.

Section 1.5 of Chapter 13 specifies key requirements for design life, operability (during normal and abnormal conditions), reliability, accessibility, maintainability, and inspectability for ALWRs referencing the passive Requirements Document.

2.2 Performance and Operational Requirements

Section 2.2 of Chapter 13 of the passive Requirements Document requires the turbine-generator design to support operation for 60 years without an extended refurbishment outage. EPRI specifies that the turbine-generator designer will identify specific actions to improve turbine-generator availability to meet the goal for overall plant average annual availability specified in Chapter 1 of the passive Requirements Document. The planned outage schedule for the ALWR turbine-generator will be concurrent with refueling outages. The overall requirements for ALWRs referencing the passive Requirements Document specify that regular, annual, planned turbine-generator maintenance will be completed within an average of 25 days or less (see Section 2 of Chapter 1 of the passive Requirements Document). EPRI indicates that an additional 180 days of outage will be allowed in a 10-year period. The turbine-generator and turbine valves should be capable of operating for a minimum of 6 years between inspections. EPRI concludes that current good practices indicate that a 10-week outage every 5 to 6 operational years is appropriate for major turbine-generator inspection.

Because several large U.S. utilities may have a substantial portion of their load served by nuclear power, some load cycling service may be desirable. Accordingly, EPRI specifies a load-follow operating cycle in Section 2.2.4 of Chapter 13.

EPRI requires that in the event of a loss of offsite power, the turbine-generator be able to operate at house loads for 2 hours without any detrimental effect on the systems or components. EPRI states that this capability will significantly reduce the frequency of a core damage accident. Section 2.2.5 of Chapter 13 requires that adequate design margin be provided to permit off-normal frequency operation without damage for periods that will

permit initiation of protective trips and recovery without trip for short-duration frequency deviations. EPRI also requires that the designs of the turbine-generator and components provide sufficient margin and redundancies to account for anticipated abnormal operating conditions, thereby avoiding limiting conditions for operation over the entire operating cycle.

2.3 System and Equipment Requirements

Section 2.3.1 of Chapter 13 of the passive Requirements Document states that the use of proven technology is fundamental to the concept of the ALWR. However, EPRI establishes no specific criteria for the number of years required before a design will be considered to be proven. EPRI states that improvements in the design of components of proven experience may be made for ALWRs referencing the passive Requirements Document without resulting in changes to the basic configuration of the component.

No code requirements are specifically applicable to the analysis of most of the major turbine-generator components. Therefore, the passive Requirements Document establishes minimum criteria for the analysis of turbine-generator components in the areas of stress, brittle fracture, fatigue, and flaw size.

EPRI establishes additional requirements for the turbine-generator system in Section 2.3 of Chapter 13 in the areas of structural foundation analysis; operability and maintainability; equipment access; maintenance, inspection, and testing requirements; modularization; and fire protection.

Section 2.3.5.2 of Chapter 13 addresses the design requirements for the structural foundation for the turbine and the generator. According to these design requirements, the plant designer will include soil-structure interactions in the foundation design and analysis and will identify the basis for allowable soil (static and dynamic) characteristics in the design analysis. In addition, the plant designer will describe the structural design parameters, the static and dynamic models used in the system analysis, and the basis for the selection of the design models and the assumed loads. The staff concludes that these design requirements for the main turbine-generator systems comply with the guidelines of SRP Section 3.7.2 and are acceptable.

3 MAIN TURBINE SYSTEM

Section 3 of Chapter 13 of the passive Requirements Document provides the definition, performance, and equipment requirements for the main turbine system. This section includes discussions about the main turbine and the following supporting subsystems: lube oil system, turbine control system, gland seal system, and their associated instrumentation.

3.1 Main Turbine

The main turbine system will include the high- and low-pressure turbines, the turbine stop and control valves, the reheat stop and intercept valves, the turning gears, and other associated piping and equipment.

3.1.1 Safety Classification

Section 3.3.1.1.2.11 of Chapter 13 of the passive Requirements Document originally required that the turbine main steam stop valve for a BWR meet seismic Category II requirements as defined in Section 4 of Chapter 1 of the passive Requirements Document. The turbine stop valve is not required to be classified as seismic Category I; however, Position C.1.e of Regulatory Guide (RG) 1.29, "Seismic Design Classification," states that the turbine stop valve should be designed to withstand (maintain its integrity) during the safe-shutdown earthquake (SSE). The original commitment to meet seismic Category II requirements was not completely acceptable. In addition, to satisfy the staff's position in Section 2.3.1.1.2 of Appendix B of this report, relative to the main steam isolation valve (MSIV) leakage pathway, (1) the piping between the turbine bypass valve and the condenser inlet, (2) the piping between the turbine stop valve and the turbine inlet, and (3) the condenser anchorage should be analyzed using a dynamic analysis to demonstrate its structural integrity and leaktightness under SSE loading conditions. In the DSER for Chapter 13, the staff reported that EPRI had committed to revise requirements in Chapters 2 and 13 to respond to these positions and identified this as a confirmatory issue. In Revision 3 of the passive Requirements Document, EPRI responded to this issue by revising Section 3.4.1.5 and adding Section 4.4.3.20 in Chapter 2 and revising Section 3.3.1.1.2.11 in Chapter 13. In Revision 5, EPRI further responded to this issue by submitting additional revisions to all of these sections.

In the revised Section 3.4.1.5 in Chapter 2 and Section 3.3.1.1.2.11 in Chapter 13, the requirements state that the turbine stop valves and the piping to the turbine casing will be included in the mathematical model for the dynamic seismic analysis of the main steam lines and branch line piping. The dynamic input loads for these analyses will be derived from a time history model analysis or an equivalent method. These analyses will demonstrate the capability of the turbine stop valve to withstand the SSE design loads combined with other appropriate loads within the limits specified for American Society of Mechanical Engineers (ASME) Code, Section III, Class 2 pipe. The staff concludes that implementation of these requirements, plus those discussed in Section 3.1.2 below, will provide reasonable assurance of the integrity of the turbine stop valve and piping to the turbine. Therefore, Position C.1.e in RG 1.29 is satisfied, the revised requirements are acceptable, and this confirmatory issue is closed.

A more detailed discussion of the staff's evaluation of the proposed revisions to Sections 3.4.1.5 and 4.4.3.20 in Chapter 2 is included in Appendix B, Section 2.3.1, "BWR Main Steamline Isolation Valves and Leakage Control/ Classification of Main Steamline of BWR," of this report.

In the DSER for Chapter 13, the staff reported that the Section 4.5.4.8 in Chapter 6 of the passive Requirements Document was not completely acceptable as a basis for demonstrating how those portions of the main turbine system that are considered a part of the MSIV leakage pathway (see Section 2.3.1 of Appendix B to Chapter 1 of this report and Chapter 3, Section 5.3.3.3 of the passive Requirements Document) will be protected against the postulated failure of non-seismic Category I items. The basis for the staff's concern is Position C.2 in RG 1.29 which addresses non-seismic Category I items whose failure could reduce the functioning of any safety-related item to an unacceptable safety level. Since this issue primarily involved the turbine-generator building, it is discussed in Section 4.4 of Chapter 6 of this report.

3.1.2 Inspection and Quality Assurance

For a BWR without the intermediate shutoff valve located in the main steam line between the outside isolation valve and the turbine stop valve, the staff concludes that the inspection and quality assurance guidelines in Standard Review Plan (SRP) Section 3.2.2, "System Quality Group Classification," Appendix A, should apply to the turbine stop valves, turbine control valves, turbine bypass valves, and the main steam leads from the turbine control valve to the turbine casing.

In the DSER for Chapter 13, the staff reported that EPRI had committed to revise Section 3.3.1.1.2.10 of Chapter 13 of the passive Requirements Document to require that all of the above items meet the quality group classifications specified in SRP Section 3.2.2, Appendix A, including the footnotes contained in Table A-1. The staff concluded that this was an acceptable requirement for the design of the ALWR passive plant and identified this as a confirmatory issue. The staff has confirmed that Revision 3 of the passive Requirements Document, Section 3.3.1.1.2.10 of Chapter 13 was revised to provide a requirement that BWR turbine stop, control, and bypass valves and the main steam leads from the turbine control valves to the turbine casing meet the quality group classifications specified in SRP Section 3.2.2, Appendix A. On the basis of this requirement and the commitment to SRP Section 3.2.2, Appendix A in Table B.1-2 of Appendix B to Chapter 1 of the passive Requirements Document, the staff concludes that all of SRP Section 3.2.2, Appendix A, including the footnotes in Table A-1 of that SRP section will be implemented in the design of the ALWR passive plant. This is an acceptable requirement; therefore, this confirmatory issue is closed.

3.1.3 Maintenance

In Section 3.2.1 of Chapter 13 of the passive Requirements Document, EPRI allows expanded operating requirements that permit on-line turbine maintenance during periods when the supporting systems are degraded. Operation in the proposed modes may create conditions that impose additional stress and harsher operating environments on the machine. For example, Section 3.2.1.4 requires the turbine system to handle a load-following duty cycle, which places added

stress on the turbine and its support systems. Section 3.2.1.5 requires that the turbine be able to operate at up to 70-percent full load with one string of either high-pressure or low-pressure feedwater heaters isolated (bypassing of heater strings is not allowed) which causes higher steam flows in the later stages of the turbine. Section 3.2.1.6 requires that the turbine be capable of long-term operation at full power with the steam reheaters out of service, which can result in the movement of the point of moisture formation in the low-pressure turbine stages. When combined, the requirements described above may necessitate the increased use of blade coatings, additional internal moisture removal devices, and additional moisture removal capacity. Although the passive Requirements Document states that the design will be able to accommodate these conditions without deleterious affects, the staff concludes that attention should be focused on the final designs and testing done by the vendor to ensure that the performance requirements are satisfactorily attained.

Section 3.3.1.2.1.1 of Chapter 13 requires that all turbine parts be accessible and capable of removal and assembly and that the vendor and plant designer jointly perform maintainability studies to ensure all major components can be efficiently removed, maintained, and reinstalled. However, this section did not discuss the turbine maintenance program. In the DSER for Chapter 13, the staff concluded that the turbine maintenance program was sufficiently important that EPRI should address the turbine rotor inspection interval in the Requirements Document. In Revision 5, EPRI added Section 2.3.8.3.2, requiring that plant designers provide recommendations for the plant maintenance program. The staff concludes that this is acceptable.

Section 2.2.3 of Chapter 13 states a design goal of a minimum of six years of operation between turbine generator inspections. This would represent a lengthening of currently-recommended inspection intervals. Although the staff identified no concerns about design improvements for the main stop and control valves and the reheat stop and intercept valves, it stated in the DSER for Chapter 13 that it needed more information before it could support this lengthening of inspection interval and identified this as an open issue. Subsequently, in Revision 5, EPRI added Section 2.3.8.3.2, requiring that plant designers provide recommendations for the plant maintenance program that will include inspection and test intervals. These recommendations would be based on both turbine missile generation probability analysis, as well as plant availability. The staff concludes that this is acceptable; therefore, this open issue is closed. The staff will review applications for FDA/DC to ensure plant designers specify an appropriate turbine inspection programs, based on a missile generation probability analysis.

3.1.4 Turbine Missiles

General Design Criterion 4 (GDC 4), "Environmental and Dynamic Effects Design Bases," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 requires that structures, systems, and components important to safety be protected against the dynamic effects that may result from equipment failures, such as the effects of turbine missiles. In support of this requirement, EPRI requires the main turbine to be designed and operated in a manner that minimizes the probability that turbine missiles will be generated.

In the DSER for Chapter 13, the staff concluded that the requirements specified in Section 3.3.1.1.1.14 for turbine orientation and placement were acceptable, pending correction of two apparent typographical errors in the probability values. The probability of missile generation for favorable turbine orientation should be 1.0E-4 per year, not 10E-4 per year, and the probability of unacceptable damage to safety-related systems and components owing to a turbine missile should be 1.0E-7 per year, not 10E-7 per year. This was identified as an open issue in the DSER. In Revision 3, EPRI corrected these errors; therefore, this DSER open issue is closed.

3.1.5 Turbine Rotor

Section 3.3.1.1.1.11 of Chapter 13 of the passive Requirements Document requires that all rotors be a one-piece design, either an integral forging or welded design (i.e., no shrunk-on disk or coupling).

The staff determined in the DSER that the use of a one-piece rotor might require that NRC add emphasis to part-machining inspections. The one-piece rotor design is a relatively new technology that only a few operating nuclear plants now use. Therefore, little operating experience is available to assess the reliability of this design.

SRP Section 10.2.3, "Turbine Disk Integrity," gives guidelines on fracture toughness, material selection, and inservice inspection, primarily for shrunk-on disk designs. Since a one-piece rotor design will be used at the ALWR, the guidelines of SRP Section 10.2.3 do not apply.

The staff concluded in the DSER for Chapter 13 that a one-piece rotor design would require stringent part-machining inspections and that the plant designer would have to provide the inspection requirements for a one-piece rotor to the NRC staff for review and approval before plant operation. In Revision 5 of the passive Requirements Document, EPRI changed Section 3.3.1.1.1.20 to Chapter 13 to require that center-boring of low-pressure turbine rotors be evaluated on a plant specific basis. The staff concludes that this is acceptable because different non-destructive examination techniques are available.

3.1.6 Performance Verification

Section 3.3.1.1.1.19 of Chapter 13 of the passive Requirements Document requires dimensional, fit-up, and assembly checks for the first production of new or significantly changed designs.

3.2 Turbine Lube Oil System

The turbine lube oil system will be designed to provide lubrication to the turbine, the generator, and the exciter bearings during all normal and abnormal conditions. In addition, the system can act as a backup to the generator hydrogen seal oil system. The requirements proposed by EPRI and specified in Chapter 13 of the passive Requirements Document for this system, notably the means for continuous or batch purification of the oil, are consistent with standard industry practice and are, therefore, acceptable.

3.3 Turbine Control System

The turbine control system, including the main stop and control valves and the reheat stop and intercept valves, will control the turbine-generator's speed, load, and overspeed. Most of the requirements placed on this system and its components in Section 3.3.3 of Chapter 13 of the passive Requirements Document are current and proven, including the requirement that sensing devices and local controllers be radiation-resistant for operation in a BWR environment. However, in Section 3.2.3.8 of Chapter 13, EPRI has relaxed the overspeed point from approximately 103 percent to 105 percent of the rated speed at which the governor would have fully closed the control and intercept valves.

Section 3.2.3 of Chapter 13 requires two independent overspeed trip devices that fully close the turbine valves at 111 percent and 112 percent of normal operating speed. Section 2.3.2.3 of Chapter 13 requires that the natural critical frequencies of the turbine generator shaft assemblies between 0 speed and 20-percent overspeed (120 percent of normal operating speed) be shown by safety analysis and testing to cause no distress to the unit during operation. Therefore, the staff concludes that the 105 percent of the rated speed for normal speed-load control is within the safety analysis limit required by EPRI and is acceptable.

Section 2.2.4.1 of Chapter 13 references Section 3.4.5 of Chapter 1, which requires that the turbine-generator be able to accommodate specific load-shed without turbine trip. Section 2.2.4.2 requires that the turbine-generator be capable of 2-hour operation with only in-house/hotel loads (offsite power not available). Current turbine control systems are not very reliable in handling designed levels of load-shed, and often the load-shed will result in a turbine trip. Additionally, the passive Requirements Document proposes operating conditions that could exist at the time of the load-shed function that might also complicate turbine control and affect the maintenance of generator voltage and frequency. EPRI or the plant designer may have to evaluate the thermal-hydraulics of the steam supply, along with the response capability of the turbine control system, to determine interactions and stability. In addition, reheaters or feedwater heaters (extraction steam) may not operate in a normal manner during load-shed. These factors could affect the capability of the turbine-generator control system to maintain the required tolerances for the electrical power supply to the station loads. EPRI states that this concern would be addressed by the turbine-generator designer during a specific phase of plant design. The staff will review the design adequacy of the turbine control system during as part of the RTNSS process during its review of an individual application for FDA/DC.

3.4 Turbine Protection System

The turbine protection system will provide for rapid closure of the main and reheat stop valves and, additionally, the control and intercept valves on receipt of a turbine trip signal. The protection system response is necessary to preclude unsafe conditions that could result in turbine damage and the generation of low trajectory missiles. Section 3.2.3 of Chapter 13 of the passive Requirements Document specifies two independent overspeed trip devices: a mechanical overspeed trip device set at 111 percent and an electric device set at 112 percent. Other diagnostic devices will provide additional trip initiation as specified in Table 13.3-1 of Chapter 13. The

staff concludes that the requirements in the passive Requirements Document provide adequate turbine protection and may improve reliability and on-line testability.

3.5 Gland Seal System

The turbine gland seal system will provide sealing steam to the annulus space where the turbine rotor shafts and large steam valve shafts penetrate their casings. The sealing portion of the system will prevent air from entering the steam, and the exhaust portion will prevent steam releases to the atmosphere. Sections 3.2.4 and 3.3.4 of Chapter 13 of the passive Requirements Document specify the requirements that EPRI proposes for this system. GDC 60, "Control of Releases of Radioactive Materials to the Environment," and GDC 64, "Monitoring Radioactivity Releases," require, in part, that releases of radioactive materials to the environment be controlled and monitored. The gland seal system must, therefore, provide for the collection and condensation of sealing steam and the venting and treatment, if necessary, of noncondensibles. The staff concludes that the requirements for controlling and monitoring the release of radioactive material associated with the turbine gland seal system are acceptable because they will serve to ensure that GDC 60 and 64 are met.

3.6 Instrumentation

Section 3.4 of Chapter 13 of the passive Requirements Document states that instrumentation will be provided for monitoring thermal, hydraulic, and electrical conditions; controlling equipment components'; and initiating alarms and automatic shutdown of the turbine-generator if an unsafe condition occurs. All monitoring instrumentation will be linked to the plant computer.

3.6.1 Turbine Supervisory Instrumentation

The turbine supervisory instrumentation listed in Table 13.3-1 of Chapter 13 will monitor thermal, hydraulic, and mechanical conditions. Section 3.4.1 of Chapter 13 requires monitoring for 26 turbine and support system parameters.

3.6.2 Alarm Initiating Devices

EPRI requires that devices for the main turbine initiate alarms in the control room for the conditions listed in Table 13.3-1 of Chapter 13. Some of these devices may also initiate turbine trip if the condition that initiates the alarm is not corrected.

3.6.3 Turbine/Reactor Interface Instrumentation

Section 3.4.3 of Chapter 13 requires a redundant device that will send a turbine trip signal to the reactor control system on sensing a turbine trip. This device will be part of the reactor protection system described in Chapter 10 of the passive Requirements Document. The staff addresses this item in Chapter 10 of this report.

3.6.4 On-Line Diagnostic Instrumentation

Section 3.4.4 of Chapter 13 states that on-line continuous monitoring and trending instrumentation will be required to increase plant availability by

detecting and locating turbine problems. At a minimum, a system using dual probes and accelerometers on the bearing pedestals will be capable of (1) diagnosing an imbalance in the rotor system, (2) locating a "wiped" bearing, and (3) locating an oil whirl/whip. EPRI states that this system can help shorten outage time by aiding in the diagnosis and correction of rotor imbalance, which has in the past caused delays at operating LWRs. EPRI recommends that designers consider including additional on-line diagnostic monitoring systems that have a continuous data collection and trending capability.

3.6.5 Performance Instrumentation

Section 3.4.5 of Chapter 13 states that the turbine will be equipped with sensing points for optional ASME performance tests. The individual utility will be allowed to select either full-scale ASME PTC-6 performance tests or reduced-scope PTC-6.1 tests.

3.6.6 Bearing Oil Drain Flow

Section 3.4.6 of Chapter 13 specifies that some means will be provided for verifying the flow of oil draining from the bearings.

3.7 Conclusion

The staff reviewed the requirements in Chapter 13 of the passive Requirements Document pertaining to the main turbine system. The staff identified several vendor- or utility-specific items that must be resolved by the plant designer, by the utility or through EPRI, before operation of a plant that references the passive Requirements Document. Except for those items identified as vendor or utility specific, the requirements are in accordance with the regulatory guidance in the applicable SRP sections and, therefore, are acceptable.

4 MAIN GENERATOR SYSTEM

Section 4 of Chapter 13 of the passive Requirements Document defines the main generator system, including performance and equipment requirements for the main generator and the excitation, stator cooling water, hydrogen cooling, hydrogen seal oil, generator hydrogen and carbon dioxide, and generator control subsystems.

4.1 Main Generator

The main generator will convert rotational energy produced by the main turbine into electrical power. Typical components of the main generator will include the generator stator frame, the stator core and windings, the generator rotor, the field windings, the collector rings and brushes, the high-voltage bushings, the bushing current transformer assembly, and the grounding system.

The interfaces that the main generator system will share with other systems are described in Section 4.1.3.1.2 of Chapter 13 of the passive Requirements Document.

4.1.1 Performance Operational Requirements

Section 4.2.1 of Chapter 13 states that the main generator will be designed to produce its rated electrical output when driven at the rated torque produced by the turbine. It will be capable of continuous operation at any loading up to the rated maximum output and will be able to withstand all expected operating transients resulting from rapid load changes and fault conditions on transmission lines.

Section 4.2.1.2 of Chapter 13 states that the generator will be designed to operate under a number of abnormal conditions. The rotor will be capable of withstanding a 20-percent overspeed without sustaining damage. The generator will be capable of operating at a reduced level of load with reduced hydrogen pressure and without stator cooling. The stator windings will be capable of operating at 130 percent of their rated armature current for 1 minute and the field (rotor) windings will be capable of operating two times a year at a field voltage of 125 percent of rated load field voltage for 1 minute.

The passive Requirements Document does not include any requirements for shop or prototype testing. Since Chapter 13 places heavy reliance on the use of turbine-generator components of proven design, the staff does not expect this to be an issue with ALWR designs that reference the passive Requirements Document.

4.1.2 Systems and Equipment Requirements

Section 2.2.1 of Chapter 13 indicates that the main generator will be designed to be highly reliable over a 60-year design life. To increase reliability, Section 4.3.1.1.2 specifies the use of improved materials, such as 18Mn-18Cr retaining rings for the rotor and proven designs for the stator and rotor cooling and insulation. The passive Requirements Document emphasizes that care must be exercised to minimize vibration and ensure adequate lubrication.

4.2 Excitation System

The excitation system will supply and control the direct current for the field winding of the generator. It will also control the voltage and reactive volt-ampere output of the main generator by controlling its excitation. Typically, components of the excitation system will include the excitation power transformer, the power rectifier assembly, the voltage regulation equipment, the field de-excitation equipment, and the exciter protection relaying and monitoring equipment.

The interfaces that the excitation system will share with other systems are described in Section 4.1.3.2.2 of Chapter 13 of the passive Requirements Document.

EPRI intends that the proposed requirements for the excitation system render that system capable of operating under a variety of abnormal operating conditions without incurring any degradation. Section 4.2.1.2 of Chapter 13 states that the requirements are the same as those for current generator designs.

EPRI provides for the use of either a static exciter, which will be designed for improved reliability and reduced maintenance, or a conventional rotating exciter.

4.3 Stator Cooling Water System

The stator cooling water system will remove heat from the generator stator and transfer it to the turbine building component cooling water system. Depending on the generator vendor's design, the stator will be cooled by either water or hydrogen. The stator cooling water system could also cool the exciter and isophase bus systems. The staff concludes that these provisions do not conflict with current regulatory requirements.

4.4 Hydrogen Cooling System

The hydrogen cooling system will transfer heat produced by ohmic and mechanical losses in the generator to the component cooling water system in the turbine building. This system will be capable of full-load operation of the generator with major active component (pumps, coolers, strainers, filters, demineralizers, etc.) out of service. Section 4.3.4 of Chapter 13 of the passive Requirements Document specifies the fouling resistance of the coolers and requires that they be constructed of materials suitable for the cooling water chemistry. The staff concludes that these provisions do not conflict with current regulatory positions.

4.5 Hydrogen Seal Oil System

The hydrogen seal oil system will prevent generator hydrogen from leaking from the shaft penetrations into the atmosphere; it also will prevent air from leaking into the hydrogen. Section 4.3.5 of Chapter 13 of the passive Requirements Document proposes that the ac-powered main seal oil pump have a dc-powered, dc-controlled pump for emergencies. Section 3.3.2 of Chapter 13 specifies that the turbine lube oil system will provide a constant source of makeup oil to the seal oil system. Further backup for this system, from a

connection to the turbine lube oil system, can be used after the hydrogen pressure has been reduced to a value lower than the lube oil pressure. In the DSER for Chapter 13, the staff expressed a concern that the continuous makeup of seal oil from the lube oil system could mask seal oil leakage into the generator until hydrogen gas contaminants are diagnosed. The staff reevaluated EPRI's requirements and concludes that the hydrogen gas purity monitor would detect such leakage before any resulting damage could occur.

4.6 Generator Hydrogen and Carbon Dioxide Systems

Sections 4.2.6 and 4.3.6 of Chapter 13 of the passive Requirements Document give the requirements proposed by EPRI for the generator hydrogen and carbon dioxide systems. These systems will maintain hydrogen pressure and purity in the generator to avoid explosive mixtures and electrical faults that could result from high moisture. They will also serve to purge the main generator with carbon dioxide. The staff concludes that the proposed performance requirements for this system do not conflict with regulatory requirements.

5 CONCLUSION

The staff concludes that the EPRI requirements established in Chapter 13 of the passive Requirements Document for the design of the main turbine-generator systems do not conflict with current regulatory guidelines and are acceptable. However, by themselves, they do not provide sufficient information for the staff to determine if the design and arrangement of plant-specific main turbine-generator systems will be adequate. Therefore, applicants referencing the passive Requirements Document will be required to demonstrate compliance with the additional guidance provided in the SRP, or provide justification for an alternative means of implementing the associated regulatory requirements.

The staff concludes that Chapter 13 specifies requirements that, if properly translated into a design in accordance with the NRC regulations in force at the time the design is submitted, should result in a nuclear power plant whose main turbine-generator system will perform as designed and have all the attributes required by the regulations to ensure no undue risk to the health and safety of the public or to the environment.

APPENDIX A
DEFINITIONS AND ACRONYMS

Appendix A of Chapter 13 of the passive Requirements Document contains definitions of terms and acronyms. The staff has provided a consolidated list of acronyms in Volume 1 of this report.

MASTER ACRONYMS LIST
EPRI PASSIVE REQUIREMENTS DOCUMENT DSER

The following is a list of acronyms and their meaning that are used throughout this report.

AAC	alternate ac
ABWR	advanced boiling water reactor
ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
A/D	analog-to-digital
ADS	automatic depressurization system
AEOD	Office for Analysis and Evaluation of Operational Data
AFW	auxiliary feedwater
AHU	air handling unit
AIF	Atomic Industrial Forum, Inc.
AISC	American Institute of Steel Construction
ALARA	as low as reasonably achievable
ALWR	advanced light water reactor
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	anticipated operational occurences
ARI	alternate rod injection
ARSAP	Advanced Reactor Severe Accident Program
ASD	adjustable speed drive
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	anticipated transient(s) without scram
AWG	American wire gauge
AWS	American Welding Society
B&W	Babcock and Wilcox
BFW	backup feedwater
BTP	branch technical position
BWG	British wire gauge
BWR	boiling water reactor
BWROG	BWR owner's group
CAE	computer aided engineering
CAGS	compressed air and gas system
CAS	central alarm station
CCDF	complementary cumulative distribution function
CCFP	conditional containment failure probability
CCIC	core coolant inventory control
CCTV	closed-circuit television
CCWS	component cooling water system
CCWS-NSSS	component cooling water system-nuclear steam supply system
CDF	core damage frequency
CDWS	chilled water system
CE	Combustion Engineering, Inc.
CET	containment event tree
CIV	containment isolation valve
CFR	Code of Federal Regulations

CMT	core makeup tank
CRA	control rod assembly
CRD	control rod drive
CRDM	control rod drive mechanism
CS	containment spray
CSS	containment spray system
CT	combustion turbine
CTSI	condensate treatment systems influent
CVCS	chemical and volume control system
CWS	circulating water system
D/A	digital-to-analog
DBA	design-basis accident
DBT	design basis tornado
DDACS	digital data acquisition and control system
DEGB	double ended guillotine break
DHR	decay heat removal
DNBR	departure from nucleate boiling ratio
DPV	depressurization valves
DS	drywell spray
DSER	draft safety evaluation report
DWST	demineralizer waste storage tank
ECCS	emergency core cooling system
EDG	emergency diesel generator
EMI	electromagnetic interference
EMS	environmental monitoring system
EOF	emergency operations facility
EOP	emergency operating procedure
EPA	electric protective assemblies
EPG	emergency procedure guideline
EPRI	Electric Power Research Institute
ERF	emergency response facility
ESF	engineered safety features
FSER	final safety evaluation report
ESFAS	engineered safety feature actuation system
ESW	essential service water
FPCCS	fuel pool cooling and cleanup system
FPLC	fission product leakage control
FPS	fire protection system
GDC	general design criterion(a)
GE	General Electric
GIMCS	generic issues management system
GIP	generic implementation procedure
GL	generic letter
GRWPS	gaseous radioactive waste processing systems
GSI	generic safety issue
HEPA	high-efficiency particulate air (filter)
HF	human factors
HPCI	high pressure coolant injection
HPI	high-pressure injection
HVAC	heating, ventilating, and air conditioning
HWC	hydrogen water chemistry
IAEA	International Atomic Energy Agency
I&C	instrumentation and controls
ICC	inadequate core cooling
ICS	integrated control system

ICST	influent to condensate storage tank
IDCOR	Industry Degraded Core Rulemaking
IEB	Inspection and Enforcement bulletin
IEEE	Institute of Electrical and Electronics Engineers
IES	Illumination Engineering Society
IGA	intergranular attack
IGSCC	intergranular stress corrosion cracking
ILRT	integrated leak rate test
IMS	information management system
IPCEA	Insulated Power Cable Engineers Association
IPE	individual plant examination
IRWST	in-containment refueling water storage tank
ISI	inservice inspection
ISLOCA	intersystem LOCA
IST	inservice testing
ITAAC	inspections, tests, analyses, and acceptance criteria
LBB	leak before break
LBHS	large bore hydraulic snubbers
LED	light emitting diode
LCO	limiting condition(s) for operation
LCS	leakage control system
LCS	local control station
LDB	licensing design basis
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LRB	licensing review branch
LRFD	load and resistance factor design
LRWP	liquid radioactive waste processing system
LTOP	low-temperature overpressure protection
LWR	light water reactor
MCC	motor control center
MCPR	minimum critical power ratio
MCR	main control room
MFW	main feed water
M-MIS	man-machine interface system(s)
MOV	motor operated valve
MSIV	main steam isolation valve
MSIVLCS	MSIV leakage control system
MTC	moderator temperature coefficient
MWD/MTV	megawatt days per metric ton uranium
MWe	megawatt electric
MWSE	makeup water systems effluent
MWSG	makeup water to steam generator
MWST	makeup water storage tank
NCC	natural convection cooldown
NFPA	National Fire Protection Association
NPHS	normal power heat sink
NPSH	net positive suction head
NRC	U. S. Nuclear Regulatory Commission
NS	non-seismic
NSSS	nuclear steam supply system
NSSFC	National Severe Storm Forecast Center
NUMARC	Nuclear Management and Resources Council
OBE	operating basis earthquake
OOA	object oriented analysis

OOD	object oriented design
OSHA	Occupational Safety and Health Administration
PAG	protective action guideline
PCC(S)	passive containment cooling (system)
PDHR(S)	passive decay heat removal (system)
PDR	NRC public document room
PGA	peak ground acceleration
PGC	power generation complex
PGP	procedures generations package
P&ID	piping and instrumentation diagram
PMF	probable maximum flood
PMP	probable maximum precipitation
PORV	power-operated relief valve
PRA	probabilistic risk assessment
PSD	power spectrum density
PSI(S)	passive safety injection (system)
PWR	pressurized water reactor
PWSCC	primary water stress corrosion cracking
QA	quality assurance
RAI	request for additional information
RCA	radiological control area
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RFI	radiofrequency interference
RFPY	reactor fuel power years
RG	regulatory guide
RHR	residual heat removal
RPS	reactor protection system
RPV	reactor pressure vessel
RSDC	reactor shutdown cooling
RVLIS	reactor vessel level instrumentation system
RWCS	reactor water cleanup system
RWCU	reactor water cleanup
SAFDL	specified acceptable fuel design limits
SAMDA	Severe Accident Mitigation Design Alternative
SAR	safety analysis report
SAS	secondary alarm station
SBO	station blackout
SBWR	simplified boiling water reactor
SCC	stress corrosion cracking
scfh	standard cubic feet per hour
SER	safety evaluation report
SFC	single failure criterion(a)
SG	steam generator
SGI	safeguards information
SGTR	steam generator tube rupture
SHARP	systematic human action reliability procedure
SI	safety injection
SLC(S)	standby liquid control system
SMB	safety margin basis
SPDS	safety parameter display system
SQUG	Seismic Qualification Utility Group
SRM	staff requirements memoranda
SROA	safety-related operator action

SRP	Standard Review Plan (NUREG-0800)
SRV	safety relief valve
SSCs	structures, systems, and components
SSE	safe shutdown earthquake
STCP	source term code package
SWS	service water system
TBCCWS	turbine building component cooling water system
TDI	Transamerica Delaval, Inc.
TMI	Three Mile Island
TMI-2	Three Mile Island Unit 2
TS	technical specification
TSC	technical support center
UBS	uniform building code
UHS	ultimate heat sink
URS	ultimate rupture strength
USI	unresolved safety issue
UT	ultrasonic test
UV	undervoltage
V&V	verification and validation
WS	wetwell spray

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