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Vol. 2, Pt. 1

NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document

Evolutionary Plant Designs
Chapter 1

Project Number 669

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

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Associate Directorate for Advanced Reactors and License Renewal
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
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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 applicable 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).

The staff of the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, has prepared Volumes 1 and 2 (Parts 1 and 2) of its safety evaluation report (SER) to document the results of its review of Volumes I and II of the Requirements Document. Volume 1, "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document - Program Summary," provides 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," gives the results of the staff's review of the 13 chapters of the Requirements Document for evolutionary plant designs. Volume 3, "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Requirements Document - Passive Plant Designs," scheduled to be issued in September 1993, will give the results of the staff's review of the 13 chapters of the Requirements Document for passive plant designs. Preliminary drafts of Volumes 1 and 2 were forwarded to the Commission and the Advisory Committee on Reactor Safeguards (ACRS) on May 12, 1992.
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 2 of this report gives the staff's regulatory analysis of those issues identified for the evolutionary plant designs. These issues have been addressed in Commission papers SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements"; SECY-91-078, "Chapter 11 of the Electric Power Research Institute's Requirements Document and

Additional Evolutionary Light Water Reactor Certification Issues"; and in draft Commission papers, "Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements," and "Design Certification and Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light Water Reactor Designs," that were issued on February 27 and July 6, 1992, respectively.

In SRM dated June 26, 1990, and April 1, 1991, the Commission provided its decisions on SECY-90-016 and SECY-91-078 as they apply to evolutionary designs. The Commission will be reviewing the basis for the approach that the staff is proposing for those issues discussed in the draft Commission papers of February 27 and July 6, 1992, and, accordingly, may at some future point in the review determine that such issues involve policy questions that the Commission may wish to consider. 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 evolutionary plant designs other than policy issues on which the staff has taken a position, but for which the Commission has not had the opportunity to provide guidance. These issues are summarized in Section 4 of Volume 1 and discussed in detail in this report.

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PREFACE

This safety evaluation report (SER) (Volume 2) documents the review by the U.S. Nuclear Regulatory Commission (NRC) staff of the 13 chapters of Volume II of the Electric Power Research Institute's (EPRI's) Advanced Light Water Reactor (ALWR) Utility Requirements Document (hereafter referred to as the "Evolutionary 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 Evolutionary 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 1350 megawatts-electric.

The design criteria specified by EPRI are intended to ensure that EPRI's policy statements discussed in Volume I 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, constructibility, 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 Evolutionary Requirements Document as closely as possible. Unless otherwise noted, references to sections of the Evolutionary Requirements Document pertain to that chapter.

Outstanding Issues

During its review of the original version of the Evolutionary 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 Evolutionary Requirements Document or (2) confirmatory issues for which the staff would ensure that EPRI met its commitments to revise the Evolutionary Requirements Document.

There are no open issues remaining on the Requirements Document for evolutionary plant designs other than policy issues on which the staff has taken a position, but for which the Commission has not had the opportunity to provide guidance. To provide continuity of the review, both the open and confirmatory items identified in the DSERs and the remaining open policy issues are listed in Section 1.4 of each chapter.

Vendor- or Utility-Specific Items

During its review of the Evolutionary Requirements Document, the staff identified items that were inadequately addressed by EPRI or were issues 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, nor 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 Sections 1.4 and 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 open policy issues and 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 managers for the staff's review of EPRI's ALWR Utility Requirements Document are J. H. Wilson and T. J. Kenyon. They 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 1, "OVERALL REQUIREMENTS"

1 INTRODUCTION

This chapter of the safety evaluation report (SER) documents the review by the staff of the U.S. Nuclear Regulatory Commission (NRC) of Chapter 1, "Overall Requirements," of Volume II of the Electric Power Research Institute's (EPRI's) Advanced Light Water Reactor [ALWR] Utility Requirements Document (hereafter referred to as the "Evolutionary Requirements Document") through Revision 3. Chapter 1 was prepared under the project direction of EPRI and the ALWR Utility Steering Committee, by ABB Combustion Engineering, Incorporated; MPR Associates, Incorporated; S. Levy, Incorporated; and TENERA, L.P.

On July 8, 1986, EPRI submitted Chapter 1 of the Evolutionary Requirements Document for staff review. By letters dated January 5 and March 18, 1987, the staff requested that EPRI supply additional information. EPRI provided the information in its response dated March 27, 1987.

On September 24, 1987, the staff issued its draft safety evaluation report (DSER) for Chapter 1 of the Evolutionary Requirements Document. On February 18, 1988, the staff issued a revision to its DSER. As a result of its review of the other chapters of the Evolutionary Requirements Document, the staff requested that EPRI provide additional information by letters dated March 25, 1988, and September 14, 1989. EPRI responded to these requests in its letters dated March 30 and December 22, 1989.

On July 12, 1990, and April 9, 1991, the staff and EPRI met with the Advisory Committee on Reactor Safeguards (ACRS) Subcommittee on Improved Light Water Reactors to discuss Chapter 1, the staff's corresponding DSER, the outstanding issues from the staff's review of Chapter 1, and EPRI's approach to resolving each issue.

On September 7, 1990, EPRI submitted Revision 1 of the Evolutionary Requirements Document. In that revision, EPRI significantly modified Chapter 1. Revisions 2, 3, and 4 of the Evolutionary Requirements Document were docketed on April 26 and November 15, 1991, and April 17, 1992, respectively. EPRI submitted additional information regarding Chapter 1 by letters dated May 13, July 2, and December 6 and 21, 1991, and January 9, February 3, and March 19, 1992.

1.1 Review Criteria

Section 1 of Volume 1 of this report describes the approach and review criteria used by the staff during its review of Chapter 1 of the Evolutionary Requirements Document.

1.2 Scope and Structure of Chapter 1

Chapter 1 of the Evolutionary Requirements Document defines the ALWR Utility Steering Committee's overall requirements for the design of nuclear power plants that have evolved from the current generation of power plants.

The key topics addressed in the Chapter 1 review include EPRI-proposed design requirements pertaining to safety, performance, structural design, materials, reliability and availability, constructibility, operability and maintainability, quality assurance, and mechanical equipment. Additional topics include EPRI-proposed requirements for licensing and the design process.

1.3 Policy Issues

In a staff requirements memoranda (SRM) dated August 24, 1987, 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 this chapter provides the staff's regulatory analysis of those issues identified for the evolutionary plant designs. These issues have been addressed in SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements"; SECY-91-078, "Chapter 11 of the Electric Power Research Institute's Requirements Document and Additional Evolutionary Light Water Reactor Certification Issues"; and a draft Commission paper, "Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements," that was issued on February 27, 1992. In its SRM dated June 26, 1990, and August 15, 1991, the Commission provided its decisions on SECY-90-016 and SECY-91-078, respectively.

The February 27, 1992, draft Commission paper has been forwarded to the ACRS. The staff will include the views of the ACRS in the final paper and document its final positions before seeking Commission approval. Since the Commission has not reviewed the approaches to resolving these issues, they do not represent agency positions.

These policy issues are considered fundamental to agency decisions on the acceptability of the ALWR designs. The policy for the evolutionary plant designs are the following. They are also listed in Table 4B.1 in Appendix B to Chapter 1 of this report.

Policy Issues for the Evolutionary Plant Designs

Policy Issue
• use of physically based source term
• anticipated transients without scram
• mid-loop operation
• station blackout
• fire protection
• intersystem loss-of-coolant accident
• hydrogen control
• core-concrete interaction - ability to cool core debris
• high-pressure core melt ejection
• containment performance
• dedicated containment vent penetration
• equipment survivability
• elimination of operating-basis earthquake

- inservice testing of pumps and valves
- industry codes and standards
- electrical distribution
- seismic hazard curves
- leak before break
- classification of main steamline of boiling-water reactor
- tornado design basis
- containment bypass
- containment leak rate testing
- postaccident sampling system
- level of detail
- prototyping
- inspections, tests, analyses, and acceptance criteria
- reliability assurance program
- site-specific risk assessments
- severe-accident mitigation design alternatives
- generic rulemaking related to FDA/DC

1.4 Outstanding Issues

The DSER for Chapter 1 of the Evolutionary Requirements Document contained the following outstanding issues:

Open Issues

(1) plant site parameters (2.3.1 and 4.5.2)

(2) classification of certain types of events (2.3.2)

(3) station blackout classification (2.3.2)

(4) EPRI ALWR public safety goal (2.3.3)

(5) 60-year design life (3.3)

(6) quality assurance program for certain seismic Category II items (4.3.1 and 4.3.2)

(7) use of Uniform Building Code Zone 2A specification (4.3.2)

(8) damping values in Code Case N-411 of American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) (4.4.3)

(9) vibratory loads with significant high-frequency input deviation from Regulatory Guide (RG) 1.92 (4.4.3)

(10) maximum ground water level (4.5.2)

(11) tornado effects - noncompliance with RG 1.76 (4.5.2)

(12) BWR safety/relief valve loads (4.5.4 and 4.5.5)

(13) leak before break (4.5.5)

(14) in-plant hazards regarding remaining BWR suppression pool loads after demonstration of leak before break (4.6.1)

(15) decoupling safe shutdown earthquake (SSE) from loss-of-coolant accident (4.6.1, 4.6.2, and Appendix B)

(16) operating-basis earthquake/SSE relationship (4.6.1, and Appendix B)

(17) seismic qualification of equipment (4.8.1)

(18) selection of materials for reactor coolant pressure boundary piping - compliance with NUREG-0313 (5.3.1)

(19) construction program quality assurance (7)

(20) reference to Institute of Electrical and Electronic Engineers (IEEE) P1023-1988 and EPRI-2360 for guidance regarding human factors engineering (8.2)

(21) meaning of NRC approval of EPRI Requirements Document (10)

Confirmatory Issues

(1) sabotage protection (2)

(2) initiating events (2.3.2)

(3) seismic ductility factors and ductility limits (4.3.2)

(4) structural codes and standards for structures, systems, and equipment (4.4, 4.4.1, and 4.4.2)

(5) compliance with General Design Criterion 4 (4.5.5)

(6) internal flooding (4.5.5)

(7) seismic and dynamic qualification by experience (4.8.1)

(8) seismically qualified anchorage (4.8.1)

(9) hardness limits for martensitic stainless steel (5.3.1)

(10) use of Alloy 600 (5.3.1)

(11) allowance for carbon and low-alloy-steel corrosion (5.3.1)

(12) failure mechanisms (6.2)

(13) construction verification milestones (7)

(14) inspectability and provisions for inservice testing (8.2 and 8.6)

(15) preventive maintenance and inspections (8.2)

(16) personnel qualification requirements (8.2)

(15) acoustical monitoring (8.2)

(18) operation problem areas (8.6)

(19) quality assurance requirements (9)

(20) quality problems during design and construction (9)

(21) updating Appendix B cross-reference table (10)

(22) cross-reference table of unresolved and generic safety issues (10)

(23) use of life extension experience (11.3)

(24) living probabilistic risk assessment (11.10)

(25) Section XI of ASME Code (12.2)

The final disposition of each of these issues is discussed in greater detail in the appropriate section of this chapter, as indicated by the parenthetical notation following each issue. Most of the issues identified in the DSER for Chapter 1 have been resolved. However, during its review of the significantly modified version of Chapter 1 of the Evolutionary Requirements Document, the staff identified two open policy issues on which the staff has taken a position, but for which the Commission has not had the opportunity to provide guidance. The open issues, with references to appropriate sections of this chapter given in parentheses, are listed below. The designators in front of each issue provide a unique identifier for each issue. The letter "E" indicates that the issue applies to evolutionary plant designs. The first number designates the chapter in which it is identified. The letter "O" designates that it is an open issue. The final number is the sequential number assigned to it in the chapter.

Open Issues

E.1.O-1 tornado wind speeds (4.5.2)
E.1.O-2 leak before break (4.5.5)

1.5 Vendor- or Utility-Specific Items

During its review of Chapter 1 of the Evolutionary Requirements Document, the staff identified items that were inadequately addressed by EPRI or were issues that could not be addressed generically. These items will have to be resolved during the staff's review or 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 (COL)).

These vendor- or utility-specific items, with references to appropriate sections of this chapter given in parentheses, are listed below. The designators in front of each issue provide a unique identifier for each issue. The letter "E" indicates that the issue applies to evolutionary plant designs. The first number designates the chapter in which it is identified. The letter "V" designates that it is a vendor- or utility-specific item. The final number is the sequential number assigned to it in the chapter.

E.1.V-1 scope of mitigation features (2.1 and 2.4)

E.1.V-2 implementation of design characteristics intended to enhance accident resistance (2.2)

E.1.V-3 bounding analysis by standard site design parameters (2.3.1)

E.1.V-4 selection of initiating events and their frequency categorization (2.3.2)

E.1.V-5 acceptance criteria for transient and accident analysis (2.3.2)

E.1.V-6 anticipated transient without scram response analysis (2.3.2)

E.1.V-7 acceptability of analytical codes and methodologies for safety analysis (2.5)

E.1.V-8 60-year plant life (3.3, 4.8.2, 8.2, and 11.3)

E.1.V-9 operation of PWR with a secured reactor coolant pump (3.5)

E.1.V-10 defense-in-depth analysis (3.5)

E.1.V-11 event response capability (3.5)

E.1.V-12 fuel burnup requirements (3.6)

E.1.V-13 extended operating life of control blades and control rod assemblies (3.6)

E.1.V-14 safety classification (4.3.1)

E.1.V-15 seismic qualification by experience (4.3.2 and 4.8.1)

E.1.V-16 non-seismic building structures (4.3.2 and 4.7.2)

E.1.V-17 structural design and construction codes (4.4 and 4.4.1)

E.1.V-18 elimination of operating-basis earthquake from design (4.4.3, 4.7.3, and Appendix B)

E.1.V-19 definition of support group (4.4.3)

E.1.V-20 use of Appendix N of ASME Code, Section III (4.4.3 and 4.7.3)

E.1.V-21 analysis of vibratory loads with significant high-frequency input (4.4.3)

E.1.V-22 use of nonlinear analysis to account for gaps between pipes and piping supports (4.4.3)

E.1.V-23 probabilistic approach for changing existing loads and/or loading combinations (4.5.1)

E.1.V-24 recurrence interval for wind loadings (4.5.2)

E.1.V-25 maximum ground water level (4.5.2)

E.1.V-26 precipitation for roof design (4.5.2)

E.1.V-27 snow loading (4.5.2)

E.1.V-28 detailed quantification of soil parameters (4.5.2)

E.1.V-29 minimum margin against liquefaction (4.5.2)

E.1.V-30 external hazards evaluation (4.5.2)

E.1.V-31 number of full-stress cycles (4.5.2 and 4.8.1)

E.1.V-32 site-specific SSE (4.5.2)

E.1.V-33 power spectrum density function of the time history (4.5.2)

E.1.V-34 external impact hazards (4.5.2)

E.1.V-35 design temperature (4.5.2)

E.1.V-38 protection against surface vehicle bombs (4.5.3)

E.1.V-39 BWR safety/relief valve loads (4.5.4)

E.1.V-40 NUREG-1061 methodology and acceptance criteria for leak before break (4.5.5)

E.1.V-41 hydrodynamic loads from safety/relief valves (4.5.5)

E.1.V-42 suppression pool dynamic loads (4.5.5)

E.1.V-43 design against internal-missile generation (4.5.5)

E.1.V-44 design of concrete containment (4.6.1)

E.1.V-45 load combinations for seismic Category I buildings and structures (4.6.1)

E.1.V-46 design of seismic Category I steel structures (4.6.1)

E.1.V-47 combination of pipe rupture loads with seismic loads for seismic Category I structures (4.6.1 and 4.6.1)

E.1.V-48 combination of loss-of-coolant-accident and SSE loads (4.6.1)

E.1.V-49 load combinations for safety-related portions of the plant (4.6.2)

E.1.V-50 dynamic analysis techniques (4.7.2)

E.1.V-51 methodology for generating design response spectra or time histories (4.7.2)

E.1.V-52 structural damping values (4.7.2)

E.1.V-53 masonry walls in Category I buildings (4.7.2)

E.1.V-54 use of expansion anchor bolts - compliance with Office of Inspection and Enforcement Bulletin 79-02 (4.7.2 and 4.7.3)

E.1.V-55 stability of shell-type structures under compression (4.7.2)

E.1.V-56 seismic evaluation and design of small-bore piping (4.7.3)

E.1.V-57 use of ASME Code Case N-411 (4.7.3)

E.1.V-58 use of ASME Code Cases N-411 and N-420 in same analysis (4.7.3)

E.1.V-59 construction of core support structures (4.7.3)

E.1.V-60 fatigue design curves (4.7.3)

E.1.V-61 use of IEEE 323 (4.8.2)

E.1.V-62 environmental qualification of mechanical and electrical equipment (4.8.2)

E.1.V-63 use of zinc to reduce radiation fields (5.2.7)

E.1.V-64 limits on nitrites, nitrates, and total halogens as chlorine (5.2.8)

E.1.V-65 grinding controls for PWRs (5.3.1)

E.1.V-66 effect of fabrication processes on intergranular stress corrosion cracking (5.3.1 and 5.3.1)

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E.1.V-73 PWR water chemistry (5.5.2)

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E.1.V-75 organizational description for reliability assurance program (6.1)

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E.1.V-77 reliability data bases (6.2)

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E.1.V-79 apportionment of contributions of structures, systems, and components to core damage frequency (6.3)

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E.1.V-85 inspection of construction activities (7 and 11.13)

E.1.V-86 quality assurance for non-safety-related facilities and systems (7)

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E.1.V-89 use of IEEE P1025 P1023-1988/D5 and EPRI-2360 for guidance regarding human factors engineering (8.2)

E.1.V-90 inspection and verification of security locks robotically (8.3)

E.1.V-91 quality assurance requirements for all equipment, structures, systems, facilities or software that have some safety importance or has one importance (9)

E.1.V-92 compliance of FDA/DC applications with Commission's regulations and guidance (10)

E.1.V-93 issue resolution for FDA/DC reviews (10)

E.1.V-94 inspections, tests, analyses, and acceptance criteria (10)

E.1.V-95 implementation of simplification objective (11.4)

E.1.V-96 implementation of standardization objective (11.5)

E.1.V-97 check valve testing methods (12.2.2)

E.1.V-98 full-flow testing of check valves (12.2.2)

E.1.V-99 qualification testing of active and nonactive motor-operated valves (MOVs) (12.2.2)

E.1.V-100 technical concerns regarding MOVs (12.2.2)

E.1.V-101 leak rate testing for individual containment isolation valves (12.2.2)

E.1.V-102 instrumentation to determine net positive suction head during all modes of operation (12.2.3)

E.1.V-103 testing of pump flow rate (12.2.3)

E.1.V-104 frequency and extent of disassembly and inspection of safety-related pumps (12.2.3)

2 SAFETY DESIGN REQUIREMENTS

In Section 2.2.F.7 of the DSER for Chapter 1, the staff identified a confirmatory issue regarding sabotage protection. EPRI had proposed to add failures caused by sabotage to its description of hazards for which the plant designer should consider spatial separation of systems and equipment as the way to protect against redundant trains being incapacitated by a single hazard. Because of the extensive revision of Chapter 1, the Evolutionary Requirements Document no longer includes this requirement. Because Section 73.55 of the Title 10 of the Code of Federal Regulations (10 CFR) does not require spatial separation of systems and equipment, the staff concludes that this DSER confirmatory issue is closed.

2.1 Introduction

The EPRI ALWR safety design requirements consist of three levels of the defense-in-depth framework, that is, resistance to accidents, prevention of core damage, and mitigation of the consequences of accidents, and are separated into two types of requirements: licensing-design-basis (LDB) and safety-margin-basis (SMB) requirements. EPRI defines the LDB to be the set of events and associated boundary conditions and assumptions that must be analyzed to satisfy regulatory requirements. This analysis will be done using conservative or appropriately justified best-estimate, NRC-approved calculation methods and assumptions and will meet NRC-mandated acceptance criteria. The Evolutionary Requirements Document states that the EPRI-defined SMB contains design requirements that go beyond the minimum required by the 10 CFR. EPRI states that these design requirements provide safety assurance beyond that provided by the regulatory requirements for investment protection and severe-accident protection.

In its letter dated April 24, 1991, the staff requested that EPRI identify additional hardware or other specific actions resulting from the SMB requirements in applicable sections of the Evolutionary Requirements Document. In its letter dated July 2, 1991, EPRI indicated that, to the extent practical, such additions of hardware or other specific actions resulting from the SMB requirements had been identified in specific sections of the Evolutionary Requirements Document. EPRI's response included some examples that satisfied this concern.

Section 2.1 of Chapter 1 of the Evolutionary Requirements Document states that safety design requirements consist of the three levels of the defense-in-depth framework. The defense-in-depth framework consists of requirements that increase the ALWR's resistance to an accident and improve its capability to prevent core damage or mitigate the consequences of an accident in the unlikely event it should progress beyond core damage.

It is unclear to the staff that EPRI's mitigation requirements are intended to include design-basis accidents (DBAs). Therefore, the staff will evaluate this matter during its review of an individual application for FDA/DC to ensure that the mitigation features of the design include consideration of both DBAs and core-damage accidents. The staff discusses mitigation of accidents further in Section 2.4 of this chapter.

2.2 Accident-Resistance Requirements

Section 2.2 of Chapter 1 of the Evolutionary Requirements Document specifies the ALWR design characteristics that are intended to enhance accident resistance, such as emphasizing simplification, providing ample design margin, using the best available materials and water chemistry, using the best proven diagnostic monitoring techniques, and maintaining a negative overall power reactivity coefficient under all conditions. Improved design margin is attained by the use of a 15-percent fuel design margin; a larger reactor vessel, pressurizer, and steam generator secondary side; and sufficient margins to limiting conditions for operation and reactor trip setpoints.

The staff concludes that these design characteristic requirements are acceptable in principle. However, it will evaluate the acceptability of the implementation of each specific design characteristic during its review of an individual application for FDA/DC.

2.3 Core-Damage-Prevention Requirements

2.3.1 General Requirements

The Evolutionary Requirements Document requires that the plant designer perform two types of analyses with respect to core-damage prevention (i.e., the LDB and the SMB analyses).

Section 2.3.1 of Chapter 1 specifies general requirements with regard to core-damage prevention. It requires the plant designer to identify a complete set of ALWR design-basis events and transients to be analyzed, taking credit for safety-grade equipment only; perform best-estimate analyses of the design-basis events to support the generation of plant operating procedures; and design the plant so as to (1) allow operators significant time to evaluate plant conditions; (2) minimize the potential for systems interactions; (3) require two separate and independent ac power connections to the grid to decrease the likelihood of a loss-of-offsite-power event; and (4) have the capability of achieving safe shutdown with safety-grade equipment only, assuming the most limiting single failure.

Because these requirements do not conflict with the Commission's regulations, the staff concludes that they are generally acceptable. However, EPRI's proposed implementation of these requirements is discussed throughout this chapter.

Physical Security

Section 2.3.1.4 of Chapter 1 of the Evolutionary Requirements Document requires the plant to be so designed that the operator has significant time before taking any action needed to prevent core damage. In its letter dated March 1, 1991, the staff asked EPRI if this additional time might warrant, for these designs, relaxing the requirement of 10 CFR 73.55(d)(7)(ii) that the access control system be designed for rapid ingress to areas containing vital equipment. In its letter dated May 13, 1991, EPRI stated that the best interest of safety lies in maximizing the available time for diagnosing the failures, assembling personnel and equipment, and implementing recovery activities, and, therefore, the access control system should still provide for rapid emergency access to vital equipment. The staff concludes that this

section is not incompatible with NRC requirements related to physical security.

Site Parameters

Among the general requirements, Section 2.3.1.8 of Chapter 1 addresses plant siting. Table 1.2-6 lists the envelope of standard site design parameters, for which the requirements are discussed in greater detail in Section 4.5.2 of Chapter 1. These siting parameters are intended to cover most, but not all, potential sites for future ALWRs in the United States. As such, the Evolutionary Requirements Document requires that the plant designer review the conditions at the plant owner's site against the standard design siting parameters in order to assess the possible need for modifying any design parameter. Further, it requires that the final design parameters to be used for the particular site be approved in writing by the plant owner.

The results of the staff's review of the envelope of standard site design parameters are given in Section 4.5.2 of this chapter. In the DSER for Chapter 1, the staff identified an open issue concerning worst-case site parameters. As for the final site design parameters to be used for any particular site, approval by the plant owner only is not sufficient. Approval by the NRC staff is also required. In addition, if one or more than one site-specific design parameter exceeds the standard site design parameters at some potential nuclear plant site, the plant owner should conduct a plant-specific evaluation against these parameters and submit a detailed review to the staff for approval.

2.3.2 Licensing-Design-Basis Requirements

Selection of Transient/Accident Events

Section 2.3.2 and Table 1.2-1 of Chapter 1 of the Evolutionary Requirements Document specify a set of event initiators to be included in LDB analyses, and the corresponding frequency categories for these events. Section 2.3.2.2 specifies that the plant designer should review and identify any additional initiating events applicable to the specific advanced plant design considering its unique design features, but limits the types of events to those historically analyzed. It also specifies that the plant designer will document the basis for selecting the frequency category for each additional initiating event and will identify potential single equipment failures that could occur coincident with the initiating events. Section 2.3.2.3 specifies that these events by themselves cannot generate more serious incidents without other incidents occurring independently. Section 2.3.2.8 also specifies that acceptance criteria for fuel, reactor pressure boundary, and containment, and offsite dose consequence limits will be presented for those additional events identified.

The event frequency categorization in the Evolutionary Requirements Document is based on that of Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis for Nuclear Power Plants," and the design requirements of American Nuclear Society (ANS) 18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants." These categories are: Condition II moderate-frequency events are those that may occur during a calendar year for a particular plant, Condition III infrequent events are those that may occur during the life of a particular plant, and Condition IV

limiting faults are those not expected to occur during the life of a plant but are postulated. NUREG-0800 (SRP) divides the events into anticipated operational occurrences (AOOs) and postulated accidents. Postulated accidents are limiting faults chosen as the design-basis accidents. AOOs are defined in 10 CFR Part 50 as those conditions of normal operation and transients that are expected to occur one or more times during the life of a plant and, therefore, encompass the moderate-frequency and infrequent events. Chapter 15 of the SRP does not specify an infrequent-incident category but does specify specific acceptance criteria for those events that can be categorized as infrequent events. The event frequency categorization specified in the Evolutionary Requirements Document is, therefore, consistent with the Commission's current licensing approach.

In Section 3.3.A.3, Table 3-2, and Table 3-3 of the DSER for Chapter 1, the staff identified concerns regarding safety analysis events, event initiators, and frequency categorization. Because EPRI has extensively revised Chapter 1, Section 2.3.2 now addresses these topics. The staff concludes that the issues discussed in the DSER are superseded by the following discussion.

Section 2.3.2 of Chapter 1 of the Evolutionary Requirements Document states that the Plant Designer shall identify additional initiating events (other than those described in Table 1.2-1) applicable to the specific advanced plant design. The staff reviewed Section 2.3.2 assuming that this table was meant to be generally appropriate for ALWR designs, and it does appear that EPRI attempted to list all of those events that are typically analyzed for the current generation of plants. However, the staff concludes Table 1.2-1 was incomplete and that the following events should be added.

Event	Frequency Category
Loss of all reactor coolant system flow	MF (moderate frequency)
Pressure regulatory failure	MF
Turbine trip without all bypass	MF

In addition, because the Evolutionary Requirements Document does not present an actual design, the staff has not found sufficient justification to reclassify the following, events or accidents.

- Rod drop accident.
- Uncontrolled rod withdrawal at power-must continue to be reviewed as a moderate-frequency event. (Table 1.2, and 1.4.2)
- Inadvertent opening of a safety or relief valve--must continue to be reviewed as a moderate-frequency event. (Table 1.2-1, 1.4 and 6.1)

Selection of the appropriate set of plant events and, in some cases, their frequency classification are specific to the design of a plant. The staff concludes that each FDA/DC applicant must justify the selection of anticipated operational occurrences and accidents. Pending modifications to the Evolutionary Requirements Document as discussed above, the staff will evaluate the acceptability of the event initiators and associated frequency categorizations selected by the designer during its review of an individual application for FDA/DC. On the basis of the above discussion, the DSER issues are closed.

Transient/Accident Acceptance Criteria

Table 1.2-2a of Chapter 1 specifies example reactor and fuel design limits and consequence analysis limits for moderate-frequency, infrequent, and limiting fault events. In Revision 3 of the Evolutionary Requirements Document, EPRI added requirements to maintain a coolable geometry and long-term cooling capability, as specified in 10 CFR 50.46, to Table 1.2-2a as part of the acceptance criteria for loss-of-coolant accidents (LOCAs).

Section 2.3.2.7 of Chapter 1 states that the plant designer will perform a consequence analysis for moderate-frequency and infrequent events with coincident single failures and specifies the acceptance criteria for these events, including limiting faults, as summarized in Table 1.2-2c. In Revision 3 of the Evolutionary Requirements Document, EPRI added a footnote to Table 1.2-2c to specify fuel cladding failure criteria for input to the radiological consequence analyses. That is, for the PWR, the fuel cladding failure criterion will be less than the 95/95 departure from nucleate boiling ratio limit, and for the BWR, it will be less than the minimum critical power ratio, except for (1) a LOCA event or (2) a fuel handling and cask drop event. For a LOCA, EPRI specifies that the vendor should use the source term as defined in Section 2.5.2 of Appendix B to Chapter 1, and for a fuel handling and cask drop event, the vendor should use the number of assemblies involved. These failure criteria are consistent with the SRP, except for the source term for LOCA consequence analysis. Table 1.2-2c also specifies limits based on 10 CFR Part 20 and Appendix I to 10 CFR Part 50 for moderate-frequency and infrequent events, and specifies that, for PWRs, the radiological consequences of infrequent events may exceed the guidelines of 10 CFR Part 20 but cannot be such that they interrupt or restrict public use of those areas beyond the exclusion areas. The staff concludes that the EPRI-proposed criteria are not specific enough to determine if they are consistent with the staff's review criteria. The plant designer should specify the exact acceptance criteria and identify deviations from those in the SRP, if any, and the bases for the deviations. The staff will address this matter during its review of an individual application for FDA/DC.

Anticipated Transients Without Scram Acceptance Criteria

Section 2.3.2.2 of Chapter 1 specifies that analysis and acceptance criteria for events involving failures of multiple active components associated with anticipated transients without scram (ATWS) and station blackout will be in accordance with 10 CFR 50.62 and 10 CFR 50.63, respectively. However, 10 CFR 50.62 does not specify analysis and acceptance criteria, except for the prescriptive equipment design requirements that were based on analyses of the current generation of LWRs.

The staff concludes that EPRI has not provided adequate requirements in Chapter 1 for analyses for ATWS events. Therefore, the staff concludes that it will be necessary for designers of ALWRs to demonstrate that the bases for which the ATWS rule was developed remain valid for their proposed design, and, therefore, the ATWS rule requirements are still appropriate. Each evolutionary LWR designer must demonstrate that the new plants will not experience ATWS behavior unexpectably more severe than that experienced in current plants. The staff will evaluate this matter during its review of an individual application for FDA/DC.

In addition, 10 CFR 50.62 requires each BWR to have a standby liquid control system (SLCS) that is automatically initiated. Previously, Section 2.5.4 of Appendix B to Chapter 1 of the Evolutionary Requirements Document required the SLCS to be manually initiated. In its December 6, 1991, letter, EPRI stated that it had determined that automatic actuation of the SLCS was appropriate for evolutionary designs and that it was modifying the Evolutionary Requirements Document to reflect that position. Revision 4 to the Evolutionary Requirements Document eliminated Section 2.5.4 of Appendix B to Chapter 1 and now required, in Chapter 5 Section 4.6.3.5.1 automatic initiation of SLCS. Therefore, this issue is closed. The staff also discusses this matter in the regulatory departure analysis in Appendix B of this chapter.

Station Blackout

In Revision 0 of the Evolutionary Requirements Document, station blackout was listed as a utility investment protection concern in Table 3-8. However, because of the importance of this issue, the staff concluded that it should be considered when the plant designer determines the most limiting event for analysis. In Section 2.3.2.2 of the revised Chapter 1, EPRI included a requirement that analysis and acceptance criteria for events involving failure of multiple active components associated with ATWS and station blackout be in accordance with 10 CFR 50.62 and 10 CFR 50.63, respectively. Also, in Table B.1-2 of Appendix B to Chapter 1, EPRI commits to meet the staff guidance in Regulatory Guide 1.155, "Station Blackout." The staff concludes that these provisions ensure that the plant designer will appropriately analyze the station blackout event in the ALWR evolutionary plant designs. Therefore, this DSER open issue is closed. Additional staff evaluation of specific EPRI requirements pertaining to station blackout in the ALWR is contained in Annex A of Appendix B to Chapter 1, Chapter 5, and Chapter 11 of this report.

2.3.3 Safety-Margin-Basis Requirements

Section 2.3.3 of Chapter 1 of the Evolutionary Requirements Document specifies that the ALWR design will be such that no fuel damage is predicted to occur as a result of a postulated near-instantaneous pipe break with an area equivalent to up to 6 inches in diameter in the reactor coolant boundary. It also requires the plant designer to perform a best-estimate analysis.

Sections 2.3.3.2 and 2.3.3.3 of Chapter 1 state that the plant will be designed so that no fuel damage will occur for at least 2 hours after a sustained loss of feedwater with no operator action (PWR only) and the core will be able to withstand a loss of offsite and onsite ac power for at least 8 hours without fuel damage. The design criteria include requirements for an independent, safety-grade onsite ac power source and a non-safety-grade, alternate ac onsite power source. Section 2.3.3.5 requires that the operator have 30 minutes or more to act to prevent damage to equipment or plant conditions resulting in a significant outage due to an accident or transient.

Section 2.3.3.6 of Chapter 1 specifies that the probabilistic risk assessment (PRA), considering both internal and external events, will confirm a mean annual core damage frequency for the design of less than or equal to 1.0E-5 per reactor-year. The Evolutionary Requirements Document also requires the plant designer to (1) fully define and document the technical basis of the PRA so that the plant owner can ensure that the reliability of risk-significant

systems, structures, and component is maintained; (2) develop the technical basis for a severe-accident management program to ensure the prevention and mitigation of core damage; and (3) use the plant-specific PRA and other relevant information to confirm that the plant design is compatible with the emergency procedure guidelines (EPGs) and severe-accident management program. This is consistent with the requirement in 10 CFR 52.47 to perform a design-specific PRA in support of an application for FDA/DC. The use of the PRA for developing and confirming the severe-accident management program and EPGs is also consistent with the Commission's severe accident policy. In its staff requirements memorandum (SRM) dated June 26, 1990, the Commission approved the use of an overall mean frequency of a large release of radioactive materials to the environment from a reactor accident that is less than 1 in 1 million per year of reactor operation. Although the current regulations do not specify requirements in numerical terms of frequency of core damage, the Commission, in its June 15, 1990, SRM, pertaining to the implementation of the NRC's safety goals, stated that "a core damage probability of less than 1 in 10,000 per year of reactor operation appears to be a very useful subsidiary benchmark in making judgments about that portion of [the NRC's] regulations which are directed toward accident prevention."

The staff concludes that the EPRI-proposed core damage frequency of 1.0E-5 per reactor-year is more restrictive than the Commission's guidance and is, therefore, acceptable. However, it will not use this goal as a staff acceptance criterion.

2.4 Mitigation Requirements

Section 2.4 of Chapter 1 of the Evolutionary Requirements Document provides the design requirements for accident mitigation, including those necessary for licensing as well as those to provide protection against severe accidents.

Section 2.4.1 of Chapter 1 provides the mitigation requirements for the licensing design basis, including requirements regarding the containment building and associated containment systems, site boundary dose criteria, source terms, and hydrogen control during degraded-core accidents. However, it does not include requirements for fission-product control or hydrogen control during design-basis LOCAs. The staff will evaluate this matter during its review of an individual application for FDA/DC to ensure that features for fission-product and hydrogen control during design-basis LOCAs are included in the design.

Section 2.4.2 of Chapter 1 provides the mitigation requirements necessary to meet EPRI's safety margin basis, including design criteria for the containment building and associated systems, source terms, and hydrogen control.

2.5 Analysis Requirements and Acceptance Criteria

Section 2.5 of Chapter 1 of the Evolutionary Requirements Document specifies the EPRI-proposed requirements for the licensing-design-basis (LDB) and safety-margin-basis analyses. EPRI states that for the LDB analysis, NRC-approved methods will be used including (1) assumptions and limits that are based on actual physical conditions during the accident or transient being analyzed with conservatism consistent with regulatory requirements and (2) acceptance criteria in accordance with NRC requirements. Analysis techniques will be proven through previous use. If changes are made to

existing techniques or new techniques are proposed, they will be identified and justified. Section 2.5.2.5 states that for events considered in the LDB analysis, acceptance criteria will be in accordance with the NRC requirements. In addition, Section 2.1 requires that only safety-grade equipment be assumed available in the LDB accident analyses, except for a limited number of multiple-failure events such as ATWS and station blackout.

These LDB requirements are consistent with NRC deterministic licensing analysis requirements. However, 10 CFR 52.47(b)(2)(i) requires that the performance of each safety feature and interdependent effects among the safety features of the ALWR designs be demonstrated and found acceptable by analysis, appropriate test programs, experience, or a combination thereof. It also requires sufficient data on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions and specified accident sequences, including equilibrium core conditions. The staff will evaluate the acceptability of the analytical codes and methodologies used for safety analyses as well as validation data, including test plans and facilities, during its review of an individual application for FDA/DC.

3 PERFORMANCE DESIGN REQUIREMENTS

3.1 Introduction

Section 3 of Chapter 1 of the Evolutionary Requirements Document provides design criteria that are not related primarily to safety or investment protection. This section includes EPRI-proposed requirements related to plant size, life, maneuvering capability, event transient response, and event response times. Design criteria for the core, radioactive waste, and onsite radiation exposure are also provided. The staff reviewed only those items that fall under the purview of its safety review.

3.2 Plant Size

EPRI states that the Evolutionary Requirements Document applies to designs extending up to 1350 MWe per unit.

3.3 Plant Design Life

Section 3.3 of Chapter 1 of the Evolutionary Requirements Document states that the plant will be designed to operate for 60 years without the need for an extended refurbishment outage. In addition, the plant will be designed to permit expeditious replacement of components because of obsolescence and failure over a lifetime of 60 years.

As stated in SECY-89-013, "Design Requirements Related to the Evolutionary Advanced Light Water Reactors," the staff will review the ALWR designs for a 60-year life notwithstanding the fact that a 40-year license term limitation is specified in the Atomic Energy Act and NRC's regulations. It is the applicants' responsibility to identify the components and systems that are affected. The staff will address plant life during its review of an individual application for FDA/DC. These applications will have to provide information and programs to support design life and the staff's review of such issues as fatigue, corrosion, and thermal aging. This item is discussed further by the staff in Section 4.7.3 of this chapter.

3.4 Maneuvering and Response to Grid Demands

Section 3.4 of Chapter 1 of the Evolutionary Requirements Document provides requirements for load following, frequency control, grid breakup, and load rejection.

3.5 Event Response Capability

Section 3.5 of Chapter 1 of the Evolutionary Requirements Document specifies the requirements pertaining to the capabilities to cope with various events.

Section 3.5.1 of Chapter 1 specifies that the plant will be designed to be capable of starting from cold shutdown and going to hot standby at full pressure and temperature in 24 hours. Similarly, EPRI requires that the reactor be capable of shutdown from the reactor critical stage at full temperature and pressure to 140 °F in 24 hours.

Section 3.5.2 of Chapter 1 requires the PWR plant to be capable of operating at reduced power with a secured coolant pump to enhance the availability of the plant and to reduce reactor trips. Appendix B to Chapter 1 indicates EPRI's commitment to comply with Generic Letter (GL) 86-09, "Technical Resolution of Generic Issue No. B-59 - (N-1) Loop Operation in BWRs and PWRs." GL 86-09 states that (N-1) loop operation is acceptable provided acceptable evaluation results are shown for certain plant-specific design characteristics, such as the impact of the down loop on instrumentation and control systems, human factors, operational systems, safety systems, status of valves, core-flow distribution, and potential for cold water reactivity insertion. Since these characteristics are highly dependent on the specific design of the plant, acceptability of operation with one secured reactor coolant pump is subject to plant-specific evaluation to address the concerns delineated in GL 86-09. The staff will evaluate that analysis during its review of an individual application for FDA/DC.

Section 3.5.3 of Chapter 1 requires that the plant responses to reactor trips, which are not complicated by failures beyond those that caused the trip, do not result (1) in the initiation of the emergency core cooling system, the primary system safety valve, or the emergency feedwater system (PWRs only) and (2) in the uncovering of the pressurizer heaters. Section 3.5.4 requires the plant to be capable of a turbine trip from 40 percent or less (BWR) and 100 percent or less (PWR) of the rated power without reactor trip and the lifting of the main steam safety valves. Section 3.5.5 requires that the loss of a running main feedwater or condensate pump while at full power not result in a reactor trip. Section 3.5.6 requires that rod insertions caused by single failures not result in a reactor trip and that the plant be capable of continued operation at reduced power. The staff considers these design objectives as important defense-in-depth goals. Each of these requirements is acceptable provided the designer performs design-specific analyses to demonstrate that the specified design limits are met. The staff will evaluate these analyses during its review of an individual application for FDA/DC.

Under the heading "Table 3-6" of the DSER for Chapter 1, the staff stated that EPRI had identified certain plant performance capabilities involving step and ramp power changes and inadvertent control rod insertion without a reactor trip. Although such activities may not impose significant challenges to fuel integrity, the staff requires that an analysis be provided to confirm that this is the case in specific plant designs. The staff will evaluate this matter during its review of an individual application for FDA/DC.

3.6 Core Performance

Section 3.6 of Chapter 1 of the Evolutionary Requirements Document requires that the ALWR core be designed for up to a 24-month fuel cycle and that fuel mechanical designs have a peak bundle burnup of at least 45,000 and 55,000 megawatt-days per metric ton of uranium (MWD/MTU) for BWRs and PWRs, respectively. These minimum fuel burnup requirements are inconsistent with the EPRI-proposed requirements of 50,000 and 60,000 MWD/MTU specified in Sections 4.2.2.2 and 7.2.2.2 of Chapter 4 for BWRs and PWRs, respectively. In addition, although these values are inconsistent with each other, they are greater than NRC-approved fuel burnup levels. To support this high fuel burnup operation, each ALWR design application will need to include sufficient

high fuel burnup data to demonstrate fuel integrity in the areas of fission gas release, cladding corrosion due to oxidation and hydriding, and reduction in cladding material strength.

In addition, Section 4.2.6.2 of Chapter 4 requires that the BWR control blades used for maximum core insertion be designed with a minimum exposure capability of 4.0E+21 neutrons/m^2 (nvt) with a target of 8.0E+21 nvt, and that the blades not used for maximum core insertion be designed for an operating life of 13 or 20 reactor full-power years (RFPYs), which may be selected by the plant owner. Section 7.2.3 of Chapter 4 requires the PWR control rod assemblies to be designed for a minimum operating lifetime of 15 RFPYs with an objective of 20 RFPYs. These requirements are beyond the operating experience data of the current LWRs. To support the desired extended operating life of the control blades and control rod assemblies, each ALWR design application will need to include sufficient performance data to demonstrate that irradiation effects, including material hardening, absorber depletion, and swelling, will not impair structural integrity.

3.7 Radioactive Waste Disposal

Section 3.7 of Chapter 1 of the Evolutionary Requirements Document specifies the onsite storage capacity for high- and low-level radioactive waste. See Chapter 12 of this report for the staff's evaluation of EPRI's requirements for radioactive waste processing and handling.

3.8 Occupational Radiation Exposure

In 1990, the average annual occupational exposure dose for U.S. nuclear power plants was approximately 340 person-rem. EPRI estimates that the ALWR can be operated so that this dose is less than 100 person-rem/year averaged over the life of the plant. To meet this ambitious goal, the Evolutionary Requirements Document lists several dose-reduction actions that should be implemented in the ALWR. EPRI states that these design changes will eliminate much of the dose incurred during nonroutine maintenance work. Improved chemistry control and selection of materials will result in reduced radiation fields, and the use of robotics will further reduce personnel exposures. These dose-reduction actions are in compliance with the guidelines of Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable," and are acceptable.

4 STRUCTURAL DESIGN BASES

4.1 Introduction

Section 4 of Chapter 1 of the Evolutionary Requirements Document includes the general structural design criteria and requirements applicable to all buildings, structures, systems, and equipment. It also addresses the passive structural requirements and requirements to ensure active equipment functions. Requirements for a unified system of classifying the structures, systems, and equipment with respect to function and structural integrity are established, as are codes and standards and acceptance criteria. Design codes and load combinations, as well as the required measures to mitigate the effects of in-plant hazards, are also established to ensure that equipment will function under specified dynamic and environmental conditions.

4.2 Relationships to Design-Basis Events

Section 4.2 of Chapter 1 of the Evolutionary Requirements Document states that the design-basis events specified in Section 2 of Chapter 1 will be used by the plant designer in implementing the design criteria in Section 4.

4.3 Classification Requirements

Section 4.3 of Chapter 1 of the Evolutionary Requirements Document provides general requirements to be used by the plant designer for safety and seismic classification of structures, systems, and equipment in the plant.

4.3.1 Safety Classification

General Design Criterion (GDC) 1, "Quality Standards and Records," of 10 CFR Part 50, Appendix A, requires that nuclear power plant structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Regulatory Guide (RG) 1.26, "Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," is the principal document used by the staff in its review of this subject. However, the Evolutionary Requirements Document proposes the use of American National Standards Institute/American Nuclear Society (ANSI/ANS) 51.1, "Nuclear Safety Criteria for the Design of Stationary PWRs," and 52.1, "Nuclear Safety Criteria for the Design of Stationary BWRs," as an alternative way of complying with RG 1.26. As discussed in Section 4.3.A of the DSER for Chapter 1, the staff has not completely endorsed these two industry standards. The standard safety analysis reports (SSARs) for Combustion Engineering. Inc.'s System 80+ and General Electric Company's advanced BWR (ABWR) Revolutionary plants also reference these standards. The staff concludes that it will complete its review of these two standard plant SSARs without these endorsements. Therefore, the staff's endorsement of these two standards is not necessary for the Evolutionary Requirements Document and any future evolutionary plant SSAR will be reviewed in a manner similar to the reviews being conducted for the System 80+ and the ABWR. Therefore, the staff will evaluate this matter during its review of an individual application for FDA/DC and this DSER open issue is closed.

4.3.2 Seismic Classification

Section 4.3.2 of Chapter 1 of the Evolutionary Requirements Document states that each of the plant structures, systems, and components will be designated as seismic Category I (C-I), seismic Category II (C-II), or non-seismic (NS). GDC 2, "Design Bases for Protection Against Natural Phenomena," in part, requires that nuclear plant structures, systems, and components important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety function. Such items are classified as seismic Category I. RG 1.29, "Seismic Design Classification," is the principal document used by the staff in its review of this subject. EPRI provides a commitment in Table B.1-2 of Appendix B to Chapter 1 to comply with RG 1.29 except for the optimization subject, BWR main steamline isolation valves and leakage control system. The staff's position on this optimization subject relative to seismic classification is provided in Section 2.3.1 of Appendix B of this chapter.

Section 4.3.2 of Chapter 1 defines the seismic Category I items as including all structures, systems, and components whose safety class is SC-2 and SC-3 as well as spent fuel pool structures, including all fuel racks. EPRI states that the seismic Category II classification will be applied to all plant structures, systems, and equipment that will not perform a nuclear safety function and whose continued function will not be required, but whose structural failure or interaction could degrade the functioning of a seismic Category I structure, system, or component to an unacceptable safety level, or could result in incapacitating injury to occupants of the control room. Non-seismic structures are those that do not fall into the seismic Category I and II definitions specified in Sections 4.3.2.1 and 4.3.2.2.1 of Chapter 1 of the Evolutionary Requirements Document.

In its letter dated April 24, 1991, the staff requested EPRI to clarify the relationship between "seismic classification" and "safety classification" and confirm the use of the term "equipment" instead of the commonly used term "components." In its July 2, 1991, response, EPRI noted that no specific relationship exists between seismic and safety classifications except that all SC-1, SC-2, and SC-3 items need to be designed to meet the C-I requirements. In addition, EPRI confirmed that the term "equipment" is synonymous with "components." The staff concludes that these responses are acceptable.

The following table summarizes the staff's understanding of the ALWR seismic classifications and their relationship to safety classifications.

ALWR Seismic Classification	Current Seismic Classification	Safety Classification	Requirements
C-I	C-I	SC-1, 2, and 3	Functional capability & structural integrity
C-II	Non-C-I	NNS*	Structural integrity (failure of structures will not affect the functions of SC-1, 2, and 3 items)
NS	Non-C-I	NNS	None

*NNS = non-nuclear safety.

Physical Security Considerations

Section 4.3.2 of Chapter 1 of the Evolutionary Requirements Document states that seismic classification will be consistent with RG 1.29. In NRC Review Guideline 17, RG 1.29 is used as a reference for determining which equipment should be protected as vital equipment in the sense of 10 CFR 73.2. In its letter of May 13, 1991, EPRI stated that this linkage between equipment designated as seismic and equipment designated as vital continues to be appropriate for this design. The staff concludes that this is consistent with current staff guidelines and is, therefore, acceptable.

Seismic Category I

Seismic Category I items include all structures, systems, and components (SSCs) designated as SC-1, SC-2, or SC-3. Seismic Category I SSCs will be designed to withstand the effects of a safe shutdown earthquake and to maintain the specified design functions.

Seismic Category II

In Section 4.3.B of the DSER for Chapter 1, the staff stated that EPRI had agreed that a quality assurance program in accordance with the applicable parts of Appendix B to 10 CFR Part 50 should be applied to seismic Category II items (i.e., structures, systems, and equipment) that will perform no safety function and whose continued operation is not required, but whose structural failure or interaction would degrade the functioning of a seismic Category I structure, system, or component to an unacceptable level. The staff also requested a specific commitment to Positions C.2 and C.4 of RG 1.29 for seismic Category II items. This commitment was added to the requirements portion of Section 4.3.2 in Revision 1 of the Evolutionary Requirements Document and is acceptable. Therefore, this DSER open issue is closed.

However, in the rationale portion of Section 4.3.2.2.1 of Chapter 1, EPRI states that extensive use of seismic qualification by experience, as addressed in Section 4.8, should expedite design efforts for seismic Category II structures, systems, and equipment. The staff's evaluation of the use of

seismic qualification by experience is provided in Section 4.8.1 of this chapter for equipment and Appendix B of this chapter for piping. In addition, the type of analyses required to satisfy Position C.2 of RG 1.29 is discussed below.

In its letter dated May 17, 1991, the staff requested that a requirement be added to Section 4.3.2.2 of Chapter 1 to state that if non-seismic Category I systems cannot be isolated from adjacent seismic Category I systems, the non-seismic system should be analyzed to the same criteria as those that are applicable to the seismic Category I system. This commitment would satisfy the guidelines in Item II.h of SRP Section 3.7.3, "Seismic Subsystem Analysis," and Item II.k of SRP Section 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment." In its letter dated August 1, 1991, EPRI stated that since there was a commitment in Table B.1-2 of Appendix B to Chapter 1 to comply with both of the above SRP sections, it was not necessary to include them as a specific requirement in this section. On the basis of the commitment in Appendix B, the staff concludes that applicable non-seismic Category I structures and systems (seismic Category II) will be analyzed using the same type of dynamic seismic analysis methodology as that used for seismic Category I structures and systems. This commitment is consistent with SRP Sections 3.7.3 and 3.9.2 and is, therefore, acceptable.

Section 4.3.2.2.2 of Chapter 1 of the Evolutionary Requirements Document states that the plant designer will select appropriate ductility factors for the seismic design of C-II components to take credit for realistic amounts of energy dissipated in such items during a seismic event. In the DSER for Chapter 1, the staff stated that limits on the ductility factors should be correlated with research results and should be reviewed and approved by the NRC. As discussed in that DSER, EPRI agreed to revise the Evolutionary Requirements Document to require the use of appropriate "approved" ductility factors. The rationale portion of this section includes the statement: "The seismic ductility factors and ductility limits selected are anticipated to consider the results of research which is documented by the time of the ALWR, and to provide significant improvement over current practice." The staff concludes that this statement resolves its concern and is acceptable. Therefore, this DSER confirmatory issue is closed.

Non-Seismic

Revision 0 of Section 4.3.B.3 of Chapter 1 did not provide specific requirements concerning the building codes to be used. This was identified as an open issue in the DSER for Chapter 1. In Revision 3 of the Evolutionary Requirements Document, Section 4.3.2.3 of Chapter 1 requires that non-seismic (NS) building structures be designed to the Zone 2A specification in the Uniform Building Code (UBC) with an importance factor of 1.25 assigned to the structures. In its letter dated April 24, 1991, the staff questioned the use of Zone 2A, which according to the UBC seismic zone map is lower than the designation for many regions in the United States. In its July 2, 1991, response, EPRI noted that the UBC Zone 2A specification is intended solely to provide a high degree of investment protection for NS items. However, since many regions in the United States are designated as UBC seismic Zone 2B or higher, the use of the Zone 2A specification in these zones may not be adequate for the design of NS items. However, as the staff has no regulatory requirements or criteria that apply to non-seismic structures, the staff's

review of an application for FDA/DC in this area will be limited to a determination that a particular structure is correctly characterized as non-seismic. On the basis of the above discussion, this DSER open issue is closed.

4.4 Codes and Standards

Section 4.4 of Chapter 1 of the Evolutionary Requirements Document provides EPRI-proposed requirements relative to the applicability of major design and construction codes, industry standards, and regulatory positions to the ALWR evolutionary plant design. Tables 1.4-1 and 1.4-3 in Chapter 1 list major structural design and construction codes that are applicable to the ALWR. Table 1.4-2 lists the industry technical standards to be used.

In the DSER for Chapter 1, the staff stated that several of these standards had not been endorsed by the staff and should not be used as the basis for plant design and construction. In Revision 3 of the Evolutionary Requirements Document, EPRI revised Section 4.4 to state that the use of applicable structural design and construction codes and industry standards that conflict with NRC positions will be resolved by the plant designer with the NRC and the resolution will be fully documented. The intent of this requirement is to ensure that the staff's review of applications for FDA/DC will be conducted using acceptance criteria that include the codes and standards most recently approved by the NRC. The staff concludes that this commitment is acceptable. Therefore, this portion of the DSER confirmatory issue is closed. The staff will evaluate this matter during its review of an individual application for FDA/DC or a combined license.

See the regulatory departure analysis in Appendix B of this chapter for additional staff discussion on the use of industry codes and standards.

4.4.1 Major Design and Construction Codes

In Revision 0 of the rationale portion of Section 4.4 of Chapter 1 of the Evolutionary Requirements Document, EPRI stated that it anticipated that the applicable edition of codes and standards will be that in effect approximately 42 months before start of construction. In the DSER for Chapter 1, the staff stated that this criterion deviated from the requirement of 10 CFR 50.55(a) that the edition and addenda of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) applied to the construction of components be determined by the provisions of Section NCA-1140 of ASME Code, Section III, incorporated by reference and rule. Paragraph NCA-1140(a)(2) states that, in no case, will the code edition and addenda dates established in the design specifications be earlier than 36 months before the date the plant construction permit is docketed. In response to this concern, EPRI deleted the 42-month criterion from the rationale portion in Revision 3 of the Evolutionary Requirements Document. In addition, in response to a concern discussed in the section entitled "Table 4-1" of the DSER for Chapter 1, EPRI added a note to Table 1.4-1 in Chapter 1 to require that all ASME Code, Section III items also satisfy the requirements of ASME Code, Section III, Divisions 1 and 2, Subsection NCA. This change complies with the 10 CFR 50.55(a) requirement and is acceptable. Therefore, this portion of the DSER confirmatory issue is closed.

In its letter dated May 17, 1991, the staff requested that EPRI revise Item g under "ASME Code" in Table 1.4-3 of Chapter 1 to add a provision that only those code cases that are approved or conditionally approved in RGs 1.84 ("Design and Fabrication Code Case Acceptability - ASME Section III, Division 1"), 1.85 ("Materials Code Case Acceptability - ASME Section III, Division 1"), or 1.147 ("Inservice Inspection Code Case Acceptability - ASME Section XI, Division 1") are available to the plant designer. In its letter dated August 1, 1991, EPRI stated that its modification of the Evolutionary Requirements Document specifying that applicable structural design and construction codes and industry standards that conflict with NRC positions will be resolved by the plant designer with the NRC (discussed in Section 4.4 above) should obviate the need for such a provision. Therefore, the staff will evaluate this matter during its review of an individual application for FDA/DC or a combined license.

4.4.2 Industry Technical Standards

Section 4.4.2 of Chapter 1 of the Evolutionary Requirements Document references Table 1.4-2, which lists the industry technical standards that will be applicable to the ALWR evolutionary plant. In the section entitled "Table 4-2" of the DSER for Chapter 1, the staff stated that since some of the standards in this table, such as American Society of Civil Engineers (ASCE) 4-86, had not been endorsed by the NRC, EPRI had agreed to revise the introduction to Section 4.4 in Chapter 1 to address this concern. As discussed in Section 4.4 above, EPRI's response is acceptable and this DSER confirmatory issue is closed.

4.4.3 Regulatory Positions

In Section 4.4.3 and Appendix B to Chapter 1 of the Evolutionary Requirements Document, EPRI identifies NRC regulatory guides and SRP sections that will be applicable to the structural design bases of the ALWR evolutionary plants. EPRI lists the applicable guidance in Table B.1-2 of Appendix B to Chapter 1. In Section 4.4.3, EPRI proposes several technical positions that, in its opinion, require exceptions to some of the regulatory positions. These issues are discussed below.

<u>Implementation of Leak-Before-Break Criteria</u>

This issue is discussed by the staff in Section 4.5.5 and Appendix B of this chapter.

<u>Damping Values</u>

Section 4.4.3.3.3 of Chapter 1 of the Evolutionary Requirements Document states that the plant designer will use approved realistic damping values in the analyses of buildings, structures, and equipment. The rationale portion of this section correctly states that RG 1.84 permits the use of damping values for piping systems in accordance with ASME Code Case N-411. The staff considers this section to be acceptable. However, the only staff-approved damping values for the design of buildings and structures are in RG 1.61, "Damping Valves for Seismic Design of Nuclear Power Plants," and RG 1.84 conditionally approves the use of Code Case N-411 for piping systems. This issue is also discussed by the staff in Section 4.7.3 of this chapter.

Elimination of Operating-Basis Earthquake From Design

In its letter dated August 1, 1991, regarding the use of a single damping value for both the operating-basis earthquake (OBE) and the safe shutdown earthquake (SSE) in the analyses of buildings, structures, and equipment, EPRI stated that it had deleted the OBE as an earthquake to be considered in the design process in Revision 2 of Section 4.4.3.3.3 of Chapter 1 and Section 2.1.1 of Appendix B to Chapter 1.

In SECY-90-016 and the draft policy paper on evolutionary and passive plants dated February 27, 1992, the staff stated that the OBE should not control the design of safety-related systems. As a result, the staff is involved in the rulemaking process for Appendix A to 10 CFR Part 100 to decouple the OBE from the SSE in siting considerations. The staff is also evaluating the possibility of redefining the OBE in order to satisfy the OBE's function without explicitly analyzing responses. This change would diminish the role of the OBE in design by establishing a level that, if exceeded, would require that the plant be shut down for inspection activities. The staff agrees in principle with EPRI regarding the deletion of the OBE from plant design. However, certain issues related to the treatment of earthquake cycles for evaluations of piping and equipment fatigue, seismic anchor motion effects, criteria on postulated pipe-break locations, and design of concrete structures need to be adequately resolved as a direct consequence of eliminating the OBE from design. The elimination of the OBE from design would require all current OBE design-related checks to be performed for the SSE. The staff is developing alternatives with the industry to revise the codes and standards when design-related checks are based on the OBE. Resolution of these issues may result in staff recommendations for changes in applicable ASME Code, Section III rules. Therefore, the staff concludes that the elimination of the OBE from design is acceptable. However, the details of how current OBE-related design checks will be performed using the SSE will be resolved between industry and the staff through the appropriate code-related activities or supplemental regulatory guidance. The supplemental regulatory guidance could be in the form of revised SRP sections or the ITAAC (inspections, tests, analyses, and acceptance criteria). The elimination of the OBE from design would require an exemption from the current regulations until the final rulemaking pertaining to Appendix A to 10 CFR Part 100 is approved. In the interim, the specification of the OBE ground motion remains an option for ALWR FDA/DC applicants. The staff will evaluate the applications of those applicants that opt to eliminate the OBE from design in accordance with the forthcoming supplemental regulatory guidance mentioned above. This is also discussed by the staff in Appendix B to this chapter.

Use of the Independent Support Motion Response Spectrum Analysis Method

In Section 4.4.C(3) of the DSER for Chapter 1, the staff stated that it did not accept the damping values in ASME Code Case N-411 for analyses that use the independent support motion response spectrum methodology. This position is reflected in the conditional acceptance of Code Case N-411 in RG 1.84, Revision 25, dated May 1988. The staff is currently accepting the above analysis technique only if the independent support motion method is defined as discussed below.

In Revision 3 of the Evolutionary Requirements Document, Section 4.4.3.3.4 of Chapter 1 EPRI states that the plant designer may use approved independent support motion response spectrum analyses techniques as a basis for seismic design and identifies this use as an exception to SRP Section 3.9.2. The staff's position regarding a definition of "approved techniques" is that this method is only acceptable when used in accordance with the information and recommendations in Sections 2.3 and 2.4 of NUREG-1061, "Report of the U.S. NRC Piping Review Committee," Volume 4. As a part of this position, a support group is defined by supports that have the same time history input. This usually means all supports located on the same floor (or portions of a floor) of a structure. The staff concludes that Sections 4.4.3.3.4 and 4.7.3.4 of Chapter 1 should be revised to provide this commitment. In the interim, the staff will review individual applications for FDA/DC in accordance with the above position. On the basis of the above discussion, the staff concludes that this issue is closed.

Use of Spectral Shifting Analyses as an Alternative to Spectrum Broadening

Sections 4.4.3.3.5 and 4.7.3.3 of Chapter 1 and Section 2.1.1 of Appendix B to Chapter 1 state that the plant designer may use spectral shifting analyses in lieu of spectrum broadening and identifies this use as an exception to SRP Section 3.9.2 and RG 1.122, "Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components." The spectral shifting procedures are described in ASME Code Case N-397. As stated by EPRI in the rationale portion of Section 4.4.3.3.5 of Chapter 1, the staff has conditionally accepted Code Case N-397 in RG 1.84. The rationale further correctly states that Code Case N-397 has been annulled and that its contents have been included in ASME Code, Section III, Subsection NCA, Appendix N. In its letter dated August 1, 1991, EPRI stated that including the above information in the rationale portion for the above sections of the Evolutionary Requirements Document was sufficient and that a change to the requirement portion of these sections was unnecessary. The staff does not agree that the plant designer will necessarily treat the rationale as a requirement. As discussed in Section 4.7.3 of this chapter, the staff has not endorsed Appendix N. Therefore, if the plant designer opts to use the spectral shifting procedures in either Code Case N-397 or Appendix N, the staff's position is that the condition in RG 1.84 will apply; that is, the staff will review the use of these procedures on a case-by-case basis. This position should be added to the requirement portion of Sections 4.4.3.3.5 and 4.7.3.3 of Chapter 1 in addition to Section 2.1.1 in Appendix B to Chapter 1. In the interim, the staff will review individual applications for FDA/DC in accordance with the above position. On the basis of the above discussion, the staff concludes that this issue is closed.

Analysis of Vibratory Loads With Significant High-Frequency Input

In Section 4.4.C(6) of the DSER for Chapter 1, the staff stated that in analyses of vibratory loads with high-frequency input, if the plant designer combines high-frequency results algebraically, this deviation from RG 1.92, "Combining Model Response and Spatial Components in Seismic Response Analysis," will have to be evaluated on a case-by-case basis. In Revision 3 of the Evolutionary Requirements Document, EPRI added a qualification to the rationale portion of Sections 4.4.3.3.6 and 4.7.3.3 of Chapter 1, and to Section 2.1.1 of Appendix B to Chapter 1, that indicates that for analyses of vibratory loads with high-frequency input, if high-frequency modal results are

combined by algebraic combination, the staff will review the methodology on a case-by-case basis. However, the staff does not agree that the plant designer will necessarily treat the rationale as a requirement. Therefore, this same qualification should be added to the requirement portion of Sections 4.4.3.3.6 and 4.7.3.3 in addition to Section 2.1.1 of Appendix B to Chapter 1. In the interim, the staff will review individual applications for FDA/DC in accordance with the above position. On the basis of the above discussion, the staff concludes that this DSER open issue is closed.

In its letter dated May 17, 1991, the staff requested that EPRI revise the requirement portion of Sections 4.4.3.3.6 and 4.7.3.12 of Chapter 1 and Section 2.1.1.2 of Appendix B to Chapter 1 to require that if nonlinear analyses are used to account for gaps between pipes and piping supports subjected to vibratory loads with high-frequency input, such analyses must be submitted to the staff for review and approval before they are used. In its response dated August 1, 1991, EPRI stated that since this procedure was identified as an exception to SRP Section 3.9.2 in the requirement portion of the above sections, no further changes were required. The staff does not agree that merely identifying this procedure as an exception to the SRP is sufficient for a requirement. The staff position applies to the requirement portion of Sections 4.4.3.3.6 and 4.7.3.12 and to Section 2.1.1.2 of Appendix B to Chapter 1. Therefore, the staff will review individual applications for FDA/DC in accordance with the above position.

Seismic Qualification of Equipment Using Experience Data

This issue is discussed by the staff in Section 4.8.1 of this chapter.

Design-Basis Tornado

The Evolutionary Requirements Document requires the use of ANSI/ANS 2.3 to define tornado effects based on an exceedance probability of 1.0E-6 per year. This represents an exception to RG 1.76, "Design Basis Tornado for Nuclear Power Plants." This issue is discussed by the staff in Section 4.5.2.5 and Appendix B of this chapter.

4.5 Design Loads and Conditions

4.5.1 Introduction

Section 4.5.1 of Chapter 1 of the Evolutionary Requirements Document provides general requirements for loads and conditions including natural phenomena, site proximity man-made hazards, plant operating loads, and in-plant hazards. Section 4.5.1.2 of Chapter 1 states that, on a case-by-case basis, the plant designer may, with the approval of the NRC, develop quantitative mechanistic design loads and combinations directly from design-basis events, using probabilistic methodology. The staff concludes that this is acceptable. However, as discussed in Section 4.5 of the DSER for Chapter 1, the staff is not accepting a probabilistic approach as a basis for changing existing loads and/or loading combinations, and the loading combinations recommended in SRP Sections 3.7, 3.8, and 3.9 remain valid. The staff will address this issue during its review of an individual application for FDA/DC.

4.5.2 Natural Phenomena

Section 4.5.2 of Chapter 1 delineates requirements for considering the effects of the natural phenomena listed in Table 1.2-6 of Chapter 1, including wind loadings, hydrology, geology and foundation conditions, earthquakes, tornadoes, and volcanic activities. This table provides the envelope of plant design parameters associated with these natural phenomena.

Wind Loadings

Revision 0 of Chapter 1 of the Evolutionary Requirements Document specified basic wind speeds of 110 mph (based on a 50-year recurrence interval) and 130 mph (based on a 100-year recurrence interval) for non-safety-related and safety-related structures, respectively. In the DSER for Chapter 1, the staff found these criteria acceptable; however, in Revision 3, EPRI changed the accepted criteria. Revision 3 specifies the same basic wind speed of 110 mph (based on a 50-year recurrence interval); however, this basic wind speed is to be scaled by an importance factor (as defined in ANSI A58.1-1982, "Minimum Design Loadings for Buildings and Other Structures,") of 1.0 and 1.11 for non-safety-related and safety-related structures, respectively.

The use of importance factor 1.11 for adjusting the recurrence interval from 50 to 100 years is suitable for the design of safety-related structures because the use of an importance factor of 1.11 to calculate the wind speed for a 100-year recurrence interval is equivalent to the guidance in SRP Section 3.3.1, "Wind Loadings." For non-safety-related structures, the use of a 1.0 importance factor implies that a 50-year recurrence interval is suitable. However, the staff's interpretation of ANSI A58.1-1982 is that an extreme wind associated with a 50-year recurrence interval is suitable to calculate the wind speed only for Categories I and IV structures. Since non-safety-related structures in an ALWR plant are more important than Category IV structures, the staff concludes that both safety-related structures and non-safety-related structures whose failure could have an adverse impact on safety-related structures should be designed for an extreme wind associated with a 100-year recurrence interval. The importance factor of 1.0 is not acceptable for non-safety-related structures that are important to safety (e.g., turbine building). EPRI has not provided adequate justification for its position. Therefore, the staff will address this item during its review of an individual application for FDA/DC.

Hydrology

Table 1.2-6 of Chapter 1 provides requirements for maximum ground water level, maximum flood level, and precipitation as follows:

- Maximum ground water level - The Evolutionary Requirements Document requires the maximum ground water level to be 2 feet below grade. This requirement is not acceptable; the maximum ground water level should be at grade. EPRI has not provided adequate justification for its position. Therefore, the staff will address this item during its review of an individual application for FDA/DC.

- Maximum flood (tsunami) level - Revision 0 of the Evolutionary Requirements Document proposed that the maximum flood level be in the range of 1 foot below to 26 feet above grade. Although in the DSER for Chapter 1, the staff concluded that this design parameter was consistent with regulatory requirements, it recommended that the Evolutionary Requirements Document specify only the upper limit, which, assuming the design level includes static water level plus wind-induced waves, should be plant grade or lower. In response, EPRI changed Table 1.2-6 to specify the maximum flood level as 1 foot below plant grade. The staff concludes that this requirement is acceptable.

- Precipitation (for roof design) - Revision 0 of the Evolutionary Requirements Document specified a maximum rainfall rate of 10 in./hour and a maximum snow load of 50 pounds per square foot (psf). In the DSER for Chapter 1, the staff stated that the rainfall rate of 10 in./hour was much too low, since the probable maximum precipitation (PMP) in a 5-minute interval over 1 mi^2 is 6.3 inches in the Great Lakes area and 6.2 inches along the Gulf Coast. Revision 1 specifies a higher 1-mi^2, 1-hour PMP of 19.4 inches, together with a 1-mi^2, 5-minute PMP of 6.2 inches. The 5-minute PMP value appears reasonable; however, it might exclude a number of sites in the Great Lakes area. The staff concludes that the plant designer should use the SRP guidelines and relevant RGs for developing an adequate structural and flood-prevention design basis for the PMP. The staff will address this item during its review of an individual application for FDA/DC. Therefore, this portion of the DSER open issue is closed.

  In the DSER for Chapter 1, the staff stated that the 50-psf snow load might limit sites to below 38° north latitude in some regions, citing, for example, the 69-psf snow load and 72-psf snow plus ice load that were used in the design of the Beaver Valley plant. However, Revision 3 retained the 50-psf snow load, without addressing the staff's concern. The staff concludes that this snow load is unacceptable and that the plant designer should use the guidelines for snow load as specified in American Society of Civil Engineers 7-88 (formerly ANSI A58.1). The staff will address this item during its review of an individual application for FDA/DC. Therefore, this portion of the DSER open issue is closed.

Geology and Foundation Conditions

Table 1.2-6 of Chapter 1 requires a minimum bearing capacity of >15 kips/ft^2 (ksf), a minimum shear wave velocity of >1000 ft/sec, and no liquefaction potential at the site-specific SSE level. In the DSER for Chapter 1, the staff concluded that these requirements were acceptable. However, in its letter dated April 24, 1991, the staff requested that EPRI clarify whether the specified minimum bearing capacity of >15 ksf was a static or a dynamic value. In its response dated July 2, 1991, EPRI stated that it was intended to be a static value. The staff finds that 15 ksf for a minimum static hearing capacity is sufficient for the expected demand from structural loading. Therefore, the staff concludes that this response is acceptable.

In its letter dated April 24, 1991, the staff indicated that Table 1.2-6 should give a range of soil properties to provide consistent guidance to the vendors of the standard plants and potential utilities. In its response dated

July 2, 1991, EPRI stated that the level of effort needed to quantify more specific soil parameters was beyond the scope of the Evolutionary Requirements Document, and that the ALWR objectives will be satisfied as long as the standard plant design will be suitable for a large range of foundation siting conditions that fall within the envelope of parameters of Table 1.2-6. The staff will address this during its review of an application for a combined license.

In its letter dated April 24, 1991, the staff requested that EPRI develop evaluation guidelines regarding the minimum margin against liquefaction. In its letter dated July 2, 1992, EPRI stated that the specific guidelines had not been developed for the Evolutionary Requirements Document and that a site-specific evaluation must be performed when a plant is to be founded on a soil site. Consistent with the scope and level of technical details in the Evolutionary Requirements Document, the staff concludes that the guidelines for minimum margin against liquefaction potential may be addressed by the applicant for a combined license as a site-specific issue if the plant is to be founded on a soil site, or if any structures are to be founded on soil having a liquefaction potential at sites with multiple soil conditions. Such guidelines should include a detailed evaluation of the liquefaction potential (as described in SRP Section 2.5.4, "Stability of Subsurface Materials and Foundations"), and consequences of liquefaction, of all subsurface soils, including the settlement of foundations. These evaluations will be based on soil properties obtained by state-of-the-art laboratory and field tests and involve application of both deterministic and probabilistic procedures.

In its letter dated April 24, 1991, the staff requested that EPRI define the analyses or evaluation methods that will be used to evaluate hazards such as active faults, man-induced hazards, and soil stability. In its response dated July 2, 1991, EPRI noted that these issues were not applicable in the design of standard plants and should be considered in site-specific assessments, and that it anticipated that NRC-approved state-of-the-art analyses and evaluation methods will be used at that time. The staff will address this issue during its review of an application for a combined license.

Earthquakes

Section 4.5.2.4 of Chapter 1 requires that the standardized seismic designs be based on the site parameter envelope of Table 1.2-6 in Chapter 1.

In Section 4.5.2.4 of Chapter 1, the operating-basis earthquake (OBE) was deleted for consideration in the design process. As discussed in Section 4.4.3 and Appendix B of this chapter, the staff is evaluating the effect of this change on current staff positions. This evaluation will address the requirement in Sections 4.5.2.4.4.1 and 4.8.1.1 of Chapter 1 that reduces the number of full-stress cycles for 1/2 safety shutdown earthquake (SSE) from 50 to 20. The results of this evaluation will be included in the supplemental regulatory guidance discussed in Section 4.4.3 of this chapter. The staff will review an individual application for FDA/DC in accordance with the supplemental guidance.

In Table 1.2.6 of Chapter 1, EPRI states that the ALWR plant design should be envelop an SSE ground motion having the following characteristics:

- peak ground acceleration (PGA) equal to 0.30g
- design response spectra in accordance with RG 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants"
- generated artificial ground motion time history to envelop the design response spectra

Regarding the SSE, Note 10 of Table 1.2-6 requires that the free field motion be specified at plant grade. Section 4.5.2.4.2 of Chapter 1 requires that the SSE spectra input motion be deconvoluted to specific points for applications to structural analysis models taking soil-structure interaction (SSI) effects into account. The requirement in Section 4.5.2.4.2 implies that the free field motion will be placed at plant grade to facilitate the deconvolution analysis. In its letter dated April 24, 1991, the staff requested that EPRI justify specifying free field motion at plant grade because SRP Section 2.5.2, "Vibratory Ground Motion," Revision 2, accepts the specification of free field motion at plant grade only when sufficient data indicate that the sites consist of relatively uniform soil or rock with smooth variation of properties with depth. When there are insufficient recorded ground motion data, or when the site is composed of one or more than one relatively thin soil layer overlying a competent foundation material, the free field motion should be specified on an outcrop or a hypothetical outcrop in the free field. For SSI analyses, Revision 2 of SRP Section 3.7.2, "Seismic System Analysis," contains similar requirements regarding the location for specifying free field motion. In its letter dated July 2, 1991, EPRI stated that the analysis of the soil structure system should follow the guidance in NRC's regulatory guides and SRP Sections, including SRP Sections 3.7.1, "Seismic Design Parameters," 3.7.2, and 3.7.3, "Seismic Subsystem Analysis." EPRI indicated that assumptions for variations in the soil properties should be accommodated in the analysis. On the basis of this response, the staff concludes that the requirement to specify the free field ground motion at plant grade is acceptable.

Although the design-basis SSE of an RG 1.60 ("Design Response Spectra for Seismic Design of Nuclear Power Plants") spectrum with a zero period acceleration of 0.30g is sufficient for most potential sites in the United States, it may not envelop the ground motion for sites near seismically active areas in the Eastern and Central United States or sites in the Western United States, in addition to those along the California coast. The staff has observed that earthquakes recorded in the Eastern United States possess more high-frequency (greater than 5 Hz) ground motion than those earthquakes whose records were used to develop the RG 1.60 response spectrum. This could limit the sites at which designs using 0.3g zero period RG 1.60 response spectrum could be located. The staff will review the site-specific SSE with respect to the design basis at the time of siting.

In Table 1.2-6, the criterion for the SSE ground motion time history (time history) is that the response spectra obtained from the time history envelop the design response spectra. In accordance with SRP Section 3.7.1, the staff's position is that this criterion should also include the requirement that the power spectrum density (PSD) function of the time history envelop an approved target PSD function if a single time history is used. In addition,

SRP Section 3.7.1 specifies a different acceptance criterion if multiple time histories are used. The staff requires that the time history comply fully with the SRP and will address this matter during its review of an individual application for FDA/DC.

Tornado

In Section 4.4.C of the DSER for Chapter 1, the staff disagreed with EPRI's proposed use of ANSI/ANS 2.3, "Standard for Estimating Tornado and Extreme Wind Characteristics at Nuclear Power Sites," to define the tornado effects for the ALWR. In its letter dated March 25, 1988, the staff issued its interim position on this issue. In Revision 3 of Section 4.5.2.5 and Table 1.2-6 of Chapter 1 and Section 2.1.2 of Appendix B to Chapter 1 of the Evolutionary Requirements Document, EPRI proposed that the maximum tornado wind speed of 260 mph and the tornado recurrence interval of 1 million years (tornado strike probability of 1.0E-6 per year) be used for the design-basis tornado. These parameters are based on ANSI/ANS 2.3. Section 2.1.2 of Appendix B to Chapter 1 of the Evolutionary Requirements Document lists tornado design for the evolutionary ALWR as a plant optimization subject. This subject is also discussed by the staff in the regulatory departure analysis in Appendix B of this chapter.

The current NRC regulatory position with regard to the design-basis tornado (DBT) is contained in two 1974 documents: WASH-1300, "Technical Basis for Interim Regional Tornado Criteria," and RG 1.76, "Design Basis Tornado for Nuclear Power Plants." WASH-1300 states that the probability of occurrence of a tornado that exceeds the DBT should be about 1.0E-7 per year per nuclear power plant, and the regulatory guide delineates the maximum wind speeds of 240 to 360 mph depending on the regions.

The staff has not endorsed ANSI/ANS 2.3. However, the regulatory positions in RG 1.76 were reevaluated by an NRC contractor using the considerable quantity of tornado data that are now available but were not when the regulatory guide was developed. The contractor's reevaluation is provided in NUREG/CR-4461, "Tornado Climatology of the Contiguous United States," dated May 1986. At the heart of this study is the tornado data tape, prepared by the National Severe Storm Forecast Center, that contains 30 years of data, 1954 through 1983. This data tape contains the data for the approximately 30,000 tornadoes that occurred during that period.

The contractor found that the tornado strike probabilities range from nearly 1.0E-7 per year for much of the Western United States to about 1.0E-3 per year in the Central United States. On the basis of discussions between the contractor and the staff, wind speed values associated with a tornado having a mean recurrence interval of 10E-7 per year were estimated to be about 200 mph for the United States west of the Rocky Mountains and 300 mph for the United States east of the Rocky Mountains.

In its December 6, 1991, letter, EPRI proposed that a maximum tornado wind speed of 300 mph and a tornado recurrence interval of 1.0E-7 per year tornado strike probability be used for the design-basis tornado in the design of the evolutionary ALWRs. In Revision 4 of Chapter 1, EPRI deleted the reference to the tornado recurrence interval in Table 1.2-6 from the Evolutionary Requirements Document.

The tornado design-basis requirements have been used in establishing structural requirements (minimum concrete wall thickness) for the protection of nuclear plant safety-related structures, systems, and components against the effects not covered explicitly in review guidance such as regulatory guides or the SRP. Specifically, some aviation (general aviation light aircraft) crashes, nearby explosions, and explosion debris or missiles have been reviewed and evaluated routinely by the staff by taking into account the tornado protection requirements. Hence, the staff's acceptance of the structural design will also necessitate a concurrent evaluation of their effect on the protection criteria for some external impact hazards, such as general aviation crashes or nearby explosions. The staff will evaluate this matter during its review of an individual application for FDA/DC or a combined license.

In the draft Commission paper on evolutionary and passive plant policy issues dated February 27, 1992, the staff stated that it will accept the tornado design basis of 300 mph recently proposed by EPRI. Table 1.2 shows the DBT parameters that the staff considers acceptable. However, until the staff resolves this issue with the Commission, it considers this an open issue that must be resolved before it can complete its review of Chapter 1.

Volcanic Activity

In its letter dated April 24, 1991, the staff requested that EPRI confirm that the ALWR site will not be located in areas subjected to the effects of volcanic activity near the site. In its July 2, 1991, response, EPRI confirmed that the standard plants will not be located in such areas and that site-specific design assessments will be necessary if a plant owner decides to locate a standard plant in such an area. The staff concludes that this response is acceptable.

Design Temperatures

Design temperatures are not included or discussed in Section 4.5.2 of Chapter 1. However, in Revision 0 of the Evolutionary Requirements Document, Table 2-1 specified three categories of design temperatures: ambient, emergency cooling water inlet, and condenser cooling water inlet. The ambient temperature was expressed in terms of the maximum and minimum temperatures for both 1-percent exceedance probability and 0-percent exceedance probability. In the DSER for Chapter 1, the staff stated that it was not certain how the ambient temperature values will be used when they are derived from a probabilistic method and are associated with certain probabilities of exceedance.

Revision 3 of Chapter 1 of the Evolutionary Requirements Document specifies the same probability-based ambient temperature values as those in Revision 0 and does not address the staff's concern. In Revision 3, the cooling water inlet temperature values specified in Revision 0 were changed. The new criterion requires that the site permit atmospheric heat rejection of cooling water system heat loads or be such as to provide cooling water at the flow rates and temperatures specified by the plant designer to achieve certain probability-based cooling performance limits. To review the safety-related water supply, the staff typically uses deterministic values based on worst 1-hour, 24-hour, and 30-day values of record. Therefore, the staff will use the deterministic approach to review an individual application for FDA/DC.

4.5.3 Site Proximity Man-Made Hazards

Physical Security Considerations

The inherent resistance to sabotage of the current generation of LWRs is in part due to reinforced-concrete structures designed to tornado design criteria. Generic Letter 89-07, "Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs," contains a safeguards information addendum that includes generic standoff distances for protection against land vehicle bombs of reinforced-concrete walls constructed to existing tornado standards. That guidance is derived from NUREG/CR-2462, "Capacity of Nuclear Power Plant Structures To Resist Blast Loadings," which in turn relies on RG 1.76. The staff questioned whether the exception taken in Section 4.4.3.3.8 of Chapter 1 to RG 1.76 would reduce the inherent resistance of an ALWR to sabotage or require different guidance on standoff distances for contingency planning. In its May 13, 1991, response, EPRI stated that the wall capacity (thickness and reinforcement) will likely be dictated by seismic shear load rather than tornado loading because of the 0.3g SSE requirement. The staff does not agree with this position. If minimum static capacities are less than those assumed in NUREG/CR-2462, the staff may need to revise its guidance on contingency planning for surface vehicle bombs. The staff will address this issue during its review of an application for a combined license.

4.5.4 Plant Operating Loads

Section 4.5.4 of Chapter 1 of the Evolutionary Requirements Document provides general requirements that state that the plant designer will consider and minimize the loads due to anticipated and extreme plant conditions resulting from the design-basis events listed in Section 2 of Chapter 1.

Anticipated plant conditions consist of normal operating loads such as live and deadweight loads, pressure, temperature effects, external reactions, and loads from expected plant transients such as plant heatup and cooldown, hot standby, and anticipated conditions during operations from zero to full power. Vibratory loads due to reciprocating and rotating equipment, pump pulsations, rapid opening and closing of valves, flow-induced vibrations, relief valve blowdown, cavitation, and any other fluid-system-induced loadings on equipment and structures will be minimized by system design, choice of equipment, and use of accumulators to dampen vibrations. Where vibrations cannot be avoided, energy absorbers, dashpots, or other such devices should be used. The use of seismic snubbers for vibration damping should be avoided. Potential water-hammer events such as turbine stop valve closure, safety/relief valve and fast valve actuations, and rupture disc loads will be considered in the design of the systems. The methodology discussed in Section 4.5.4.1.5 of Chapter 1, for defining BWR safety/relief valve loads to be used in the design of the suppression pool is still under review by the staff. The staff will evaluate this matter during its review of an individual application for FDA/DC.

Extreme plant condition loads include pressures, temperatures, forces, and vibrations due to transients and accidents not encountered in normal operation, but that might reasonably be anticipated to occur during the design life of the plant. Such loads include, but are not limited to, the following: (1) water hammer in subcooled water lines or flooded steam lines; (2) abnormal component operation, such as a relief valve stuck open or an isolation valve stuck closed downstream of a pump; and (3) low-flow thermal stratification in

feedwater lines. The plant designer will minimize the effects of such loads by designs that reduce or isolate the loads and by accounting for the loads early in the plant design.

The staff concludes that, with the exception noted above, the broad requirements in Section 4.5.4 of Chapter 1 relative to plant operating loads are consistent with current applicable staff guidelines and are acceptable.

4.5.5 In-Plant Hazards

Section 4.5.5 of Chapter 1 states that the plant designer will consider the potential structural effects of in-plant hazards due to the environmental conditions associated with plant operations under normal conditions and postulated accidents. These hazards are pipe rupture, internal flooding and fires, and the generation of missiles.

Rupture of Piping

Section 4.5.5.1 of Chapter 1 provides general requirements to be used by the plant designer for determining postulated pipe-rupture locations and the dynamic effects associated with the postulated rupture of piping. The criteria of ANSI/ANS 58.2, "Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Pipe Rupture," as supplemented by applicable regulatory positions, will be used for these evaluations. The dynamic effects that are to be evaluated include impact from whipping pipes, jet impingement loads, rapid subcompartment pressurization, hydraulic system internal loads, and the motion of the equipment attached to components responding to these effects. Those portions of safety-related structures, systems, and components that could be affected by these dynamic effects will either be protected from such effects or be designed to withstand the resulting loads and still be able to perform the required safety function. Table B.1-2 of Appendix B to Chapter 1 commits to compliance with SRP Section 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated With the Postulated Rupture of Piping," with no exceptions. The staff has reviewed ANSI/ANS 58.2 and concludes that it is consistent with SRP Section 3.6.2. The staff further concludes that the commitment to ANSI/ANS 58.2, supplemented by applicable positions in SRP Section 3.6.2, provides acceptable criteria for determining postulated pipe-rupture locations and associated dynamic effects.

Pipe Rupture Loads

The requirements regarding pipe rupture loads in Section 4.5.5 of Chapter 1 of the Evolutionary Requirements Document contain a statement that subsets of safety class structures, systems, and components shall be protected from the dynamic effects resulting from postulated ruptures or designed for the resulting loads. The dynamic effects that will be considered in the design include pipe whip, jet impingement, rapid subcompartment pressurization, hydraulic system internal loads, and the motion of attached equipment. It is the staff's understanding that this requirement is not applicable if the ALWR applicant has justified the use of the leak-before-break (LBB) concept. To be consistent with the implementation of the LBB technology under the recent broad-scope amendment of GDC 4 of Appendix A to 10 CFR Part 50, containment and emergency core cooling system (ECCS) functional and performance requirements are maintained. The staff's current interpretation of this rule change

is that the above local dynamic effects (pipe whip, jet impingement, etc.) can be excluded from the containment design basis. To retain high safety margins, the containment must continue to be designed to withstand all global loading and environmental effects up to and including the double-ended rupture of the largest pipe in the reactor coolant system.

Leak Before Break

In Appendix A to the DSER for Chapter 1, the staff identified leak-before-break (LBB) considerations as an open issue. Compliance with General Design Criterion (GDC) 4 of Appendix A to 10 CFR Part 50 was listed as a confirmatory issue in Section 10 of the DSER. Revision 3 of Section 4.5.5 of Chapter 1 of the Evolutionary Requirements Document specifies that, to the extent practicable, fracture mechanics and mechanistic analysis through the application of LBB criteria will be used to minimize the need to evaluate the effects of pipe rupture. The LBB approach is also addressed in Sections 4.4.3.3.1 and 4.5.5.1.3 of Chapter 1 of the Evolutionary Requirements Document.

EPRI is proposing to adopt for ALWRs the LBB approach when certain details of the piping design, materials properties, and stress conditions are known. The regulatory departure analysis in Appendix B of this chapter provides the background and staff position on this issue. In the draft Commission paper on evolutionary and passive plant policy issues dated February 27, 1992, the staff concludes that the application of the LBB approach to ALWRs for which FDA/DC is being sought under 10 CFR Part 52 is acceptable when appropriate bounding limits are established during the FDA/DC review phase using preliminary analyses results and are verified during the combined license phase by implementing the appropriate inspections, tests, analyses, criteria, and acceptance criteria (ITAAC) discussed herein. Since the Commission has not yet reviewed this approach to resolving the approach for resolving leak before break, it does not represent an agency position. Therefore, the staff regards this as a open issue that will be closed once the Commission approves this resolution or provides alternative guidance.

The staff has evaluated the Evolutionary Requirements Document for LBB applications in ALWRs and concludes that it is not completely acceptable. The limitations and acceptance criteria for LBB applications in ALWRs are the same as those established for currently operating nuclear power plants. Therefore, NUREG-1061, Volume 3, should be referenced by the individual FDA/DC or combined license applicant requesting approval of the LBB approach because it provides the NRC-approved LBB methodology and acceptance criteria. In the rationale portion of Section 4.5.5.2.4, EPRI states that its LBB methodology and acceptance criteria are the same as those in NUREG-1061, Volume 3. However, the staff has determined that they are not the same because the criteria in Section 4.5.5.2.4 are merely excerpts of the acceptance criteria in NUREG-1061. The staff will evaluate this matter during its review of an individual application for FDA/DC or a combined license.

BWR Suppression Pool Loads

In Appendix A to the DSER for Chapter 1, the staff stated that it had not completed its evaluation of the methodology proposed by EPRI for defining BWR safety/relief (SRV) valve loads to be used in the design of the suppression pool. The methodology proposed by EPRI to be imposed on the submerged structures within the suppression pool during SRV discharge did not

specifically reference staff-approved methods or propose verification of calculational techniques if new methods are used. For the Mark I, II and III BWR containment designs, a substantive test program was undertaken to develop a load definition report (LDR) that established both acceptable models as well as the basis for NRC acceptance of these models to calculate suppression pool hydrodynamic loads.

However, if either the methodology or the data base is used, the FDA/DC applicant must demonstrate applicability to the specific design. Parameters that could influence applicability include, but are not limited to, the following: quencher design details, piping layouts, submergence, piping diameter, and safety valve lift pressure. In addition to the physical layout, the response of the primary system is also a factor. This response is critical because it will determine the number of valves that will lift and the number that will reopen on the "second pop." Because of the necessary detail associated with this issue, the staff will conduct a specific design review of plant hydrodynamic loads and will evaluate the method of determining the SRV loads and the applicability of the data base used to support the model. Should the LDR be sufficiently different from the specific design, a new plant-specific experimental basis similar to the work done at the full-scale test facility would need to be developed before staff acceptance.

The staff concludes that EPRI has not justified its position for the use of a model to compute SRV loads. The staff will review individual applications for FDA/DC against the criteria in the SRP. On the basis of the above discussion, this DSER open issue is closed.

In Section 4.5.5.3.1 of Chapter 1, EPRI states that the models used to compute pool swell, condensation oscillation loads, and chugging loads are assumed applicable for calculating the containment boundary loads. The staff concludes that this is acceptable if the designer can demonstrate that the design is within the limits of the data base.

The Evolutionary Requirements Document does not address the methodology or data base to be used for calculating the hydrodynamic loads. Because the configuration of the ABWR vent and containment appears to be a combination of the Mark II and Mark III configurations, the methodology used by the plant designer must be justified. The plant designer must also show the applicability of the data base to the design. The following critical variables may have an effect on pool loads: vent diameter and vent pipe length, submergence, pool temperature limits, containment volume, and configuration. These variables need to be addressed by the plant designer. Because of the necessary detail associated with this issue, the staff will conduct a specific design review of plant hydrodynamic loads and will evaluate the method of determining the loss-of-coolant-accident (LOCA) loads and the applicability of the data base used to support the model. Should the Mark III data base be sufficiently different from the specific design, a new plant-specific experimental basis similar to the work done to establish the generic methodology for establishing design-basis LOCA loads for the Mark II and III designs would need to be developed before staff acceptance.

The staff concludes that EPRI has not completely justified its position for the use of the ULD models to compute pool dynamic loads. The staff will evaluate this matter during its review of an individual application for FDA/DC.

The heat removal and mass replacement capability of the ECCS (flow rates, pressures, storage volumes) should continue to be designed to accommodate pipe ruptures up to and including the double-ended rupture of the largest pipe in the reactor coolant system, even when LBB is demonstrated.

Internally Generated Missiles

In Table B.1-2 of Appendix B to Chapter 1, EPRI commits to comply with the staff review guidance in SRP Section 3.5.1.1, "Internally Generated Missiles (Outside Containment)"; SRP Section 3.5.1.2, "Internally Generated Missiles (Inside Containment)"; and SRP Section 3.5.1.4, "Missiles Generated by Natural Phenomena." The staff concludes that this commitment is acceptable. However, Section 4.5.5.4.1 of Chapter 1 states that ANSI/ANS 58.1, "Plant Design Against Missiles," will be used for guidance in meeting the requirements pertaining to internal-missile generation. The staff has not endorsed ANSI/ANS 58.1. Therefore, if differences exist between the above SRP sections and ANSI/ANS 58.1, the SRP sections should be used. If a plant designer identifies and provides justification for the differences, the staff will review the justification on a case-by-case basis and address the issue during its review of an individual application for FDA/DC.

Internal Flooding

In Section 2.2.F.7 of the DSER for Chapter 1, the staff recommended that EPRI add information on protection against and mitigation of internal flooding. In Table B.1-2 of Appendix B to Chapter 1 of the Evolutionary Requirements Document, EPRI commits to comply with SRP Section 3.4.1, "Flood Protection." Section 4.5.5.5 of Chapter 1 states that the plant designer will identify those systems and equipment that must be protected from flooding and those that must be capable of normal operation under flooded conditions. The specific design conditions will be specified for systems and equipment required to operate when directly exposed to flooded conditions. EPRI states that internal flooding may be caused by such events as fire protection system operation or postulated breaks in tanks or piping. In addition, EPRI added information about internal flooding to Table 1.2-4 of Chapter 1 in Revision 4 of the Evolutionary Requirements Document. The staff concludes that the above commitment and the requirements in Section 4.5.5.5 and Table 1.2-4 provide the necessary requirements to protect structures and systems from the effects of internal flooding and are, therefore, acceptable.

4.6 Load Combinations

4.6.1 Buildings and Structures

Section 4.6.1 of Chapter 1 of the Evolutionary Requirements Document delineates the load combination requirements for the design of buildings and structures.

Concrete and Steel Containments

Section 4.6.1.1 of Chapter 1 requires that the design of concrete containments satisfy the load combinations in ASME Code, Section III, Division 2, Subsection CC, and the design of steel containments follow SRP Section 3.8.2, "Steel Containment." The staff concludes that the specified load combinations for the steel containment design conform with SRP Section 3.8.2 and are

acceptable. Because SRP Section 3.8.1, "Concrete Containment," and RG 1.136, "Materials, Construction, and Testing of Concrete Containment," provide additional guidance on the use of ASME Code, Section III, Division 2, Subsection CC, for the design of concrete containments but are not referenced in the Evolutionary Requirements Document, in a letter dated April 24, 1991, the staff requested that EPRI confirm its position regarding compliance. In its letter dated July 2, 1991, EPRI indicated that although it is impractical to list all applicable regulatory guides and SRP sections in individual paragraphs of the Evolutionary Requirements Document, the requirement of compliance for the ALWRs is shown in Table B.1-1 of Appendix B to Chapter 1. To ensure that the plant designer will use proper additional regulatory guidance for the load combinations, the staff's position is that the concrete containment design will follow the guidelines of SRP Section 3.8.1 and RG 1.136. The staff will evaluate this issue during its review of an individual application for FDA/DC.

Other Seismic Category I Buildings and Structures

Section 4.6.1.2 of Chapter 1 requires the design of other seismic Category I reinforced-concrete and steel structures to satisfy the load combinations specified in American National Standards Institute/American Concrete Society (ANSI/ACI) 349, "Code Requirements for Nuclear Safety-Related Structures," and ANSI/American Institute of Steel Construction (AISC) N690, "Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Power Plants," respectively. In its letter dated April 24, 1991, the staff noted that RG 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants," and SRP Sections 3.8.3, "Concrete and Steel Internal Structures of Steel or Concrete Containments," and 3.8.4, "Other Seismic Category I Structures," provide additional guidance on the use of ANSI/ACI 349 for seismic Category I concrete buildings and that the NRC has not approved the use of ANSI/AISC N690 for seismic Category I steel structures. In its response of July 2, 1992, EPRI indicated that although it is impractical to list all applicable regulatory guides and SRP sections in individual paragraphs of the Evolutionary Requirements Document, the requirement of compliance for the ALWRs is shown in Table B.1-1 of Appendix B to Chapter 1. To ensure that the plant designer will use proper additional regulatory guidance for load combinations, the staff position is to require adherence to SRP Sections 3.8.3 and 3.8.4 and RG 1.142. The staff will evaluate compliance during its review of an individual application for FDA/DC.

EPRI proposes to use ANSI/AISC N690 for the design of seismic Category I steel structures. The acceptability of using this code is uncertain because it has not been reviewed and approved by the staff. Therefore, the staff will evaluate this issue during its review of an individual application for FDA/DC.

Decoupling of Safe Shutdown Earthquake and Pipe Rupture Loads for Buildings and Structures

In Appendix A to the DSER for Chapter 1, the staff indicated that it was still evaluating EPRI's proposal to decouple the loads from a safe shutdown earthquake (SSE) and LOCA when the leak-before-break (LBB) approach is used. In addition, the staff raised a concern regarding protection of containment components and emergency core cooling system (ECCS) hardware against the dynamic effects of pipe ruptures for systems for which the LBB approach is used.

Section 4.6.1.3 of Chapter 1 of the Evolutionary Requirements Document specifies that when the LBB approach can be demonstrated to apply, SSE and pipe rupture loads will not be combined in the design of structures. In its letter dated April 24, 1991, the staff recommended use of the present practice of combining the loads for the design of the containment structure, structures inside the containment, and structures integral with the containment. In its response of July 2, 1991, EPRI maintained its original position as stated above.

In Section 4.6.1.1 of Chapter 1, EPRI requires using load combinations as specified in SRP Section 3.8.2 for steel containment structures and in ASME Code, Section III, Division 2, Subsection CC, for concrete containment structures. The only exception is that the Evolutionary Requirements Document excludes the operating-basis-earthquake (OBE) load terms from load combinations. The staff generally agrees with the approach to decouple the OBE from the SSE and possibly eliminate the OBE altogether from the design as discussed in Section 4.4.3 of this chapter.

In addition, in Section 4.6.1.3 of Chapter 1, EPRI proposes eliminating the combination of pipe rupture loads with seismic loads for buildings and structures when the LBB approach applies. SRP Section 3.8.2 and Table CC-3230-1 of ASME Code, Section III, Division 2, Subsection CC, require that SSE and LOCA loads be combined for containment structural design. Furthermore, EPRI proposes in Section 4.6.1.7.1 of Chapter 1 to invoke a probability-based load combination with a cutoff frequency of 1.0E-6, on the basis of ANSI/ANS 2.12 and 51.1 for the design of the containment and other seismic Category I structures. In Section 4.6.2.3 of Chapter 1, EPRI specifies that the combination of LOCA and SSE loads for equipment and systems should explicitly be eliminated on the basis of the recommendations of NUREG-1061 and implementation of the LBB technology. EPRI states that the probability of a seismically induced LOCA is extremely low for PWR primary systems and for the BWR as well.

In the early 1960s, the double-ended guillotine break of reactor coolant loop piping was postulated for containment sizing and ECCS performance. Later, this pipe rupture load was combined with the earthquake loads and applied to the containment structural design and subsequently to the design of other plant features, including nuclear reactor piping and its support systems. Since the early 1970s, the NRC criteria for design and analysis of seismic Category I structures have been formulated with sufficient conservatism to ensure ample safety margins against premature failures. The margins that were built into the structures have served as one important element in the NRC staff's implementation of the defense-in-depth regulatory philosophy, and often provided key bases for the staff to justify and allow continued operation of existing plant structures whenever load demand for the structures had to be increased for a variety of reasons (e.g., increased hazards, design errors, or omissions and modifications).

The NRC promulgated a rule change to General Design Criterion 4 of 10 CFR Part 50, Appendix A, that allows the application of the LBB method to piping systems. The revision eliminates the need to design for the dynamic effects of postulated pipe breaks, including pipe whip restraints and jet impingement shields, and to design subcompartments for dynamic pressurization loads if these loads are not essential to the containment function. Also, when the LBB

method is demonstrated deterministically applicable to a piping system, the combination of SSE loads and LOCA loads effectively becomes the SSE loads alone.

Although it has been considering modifications to SRP Sections 3.8.1 through 3.8.4, the staff found that there was an insufficient technical basis to extend decoupling to structures without resolving questions about the long-term, intermediate-term, and short-term pipe rupture effects of decoupling the pipe rupture from seismic loads.

The application of LBB technology eliminates the local dynamic effects of postulated pipe ruptures from the design basis. However, global effects still result from a source other than the postulated pipe rupture. Because the global effects from the postulated pipe rupture provide a convenient and conservative design envelope, and the NRC staff is not prepared at this time to propose alternative criteria for the containment, the load combinations indicated in the relevant sections of the SRP continue to be valid except for the combinations with the OBE. The staff will require consideration of certain aspects of the OBE in the design process to account for low-cycle fatigue and relative support motion. Although an adequate basis exists for the above-proposed decoupling of LOCA and SSE loads and the consequent deviation from the current NRC criteria for the dynamic effects for the mechanical design of components and their supports, this deviation would result in a significant reduction in critical structural safety margins for ALWR containments vulnerable to severe-accident loadings. On the basis of its understanding of the design of all the existing containment structures, the staff concludes that the containment and other seismic Category I structures must continue to contain a concurrent application of LOCA and SSE loads. For the design of seismic Category I structures, elimination of load combinations with a probability of occurrence less than 1.0E-6 per reactor-year (for other than LOCA plus SSE) is acceptable in principle; however, the staff will only approve such an elimination when a specific design applicant justifies its position with specific examples of how and where the design is governed and a comparison of structural margins with and without the load combination in question.

In summary, the staff concludes that eliminating the combination of pipe rupture loads (global effects) with seismic loads for the containment and other seismic Category I structures is not acceptable. Furthermore, EPRI's proposal to decouple LOCA and SSE loads for equipment and systems is not acceptable at this time because of the insufficient technical bases to extend the decoupling to structures. The staff will evaluate this issue during its review of an individual application for FDA/DC.

Treatment of Loads That Reduce the Effects of Other Loads

Section 4.6.1.4 of Chapter 1 requires that consideration be given to the treatment of loads that reduce the effects of other loads in the application of the design loading combination of seismic Category I buildings and structures. If any such load reduces the effects of other loads, the corresponding coefficient for that load will be taken as 0.9 if it can be demonstrated that the load is always present or occurs simultaneously with the other load. Otherwise, the coefficient for the load will be taken as zero. The staff disagrees with this criterion. EPRI is unclear about the use of a coefficient of 0.9 or 0.0 in the context of a working stress design or load factor design.

The load combinations identified in SRP Section 3.8.3 represent current practice and are appropriate for the design to ensure adequate margins of safety. It is inappropriate to use a load coefficient of 0.0. If the load is zero, its effects will be reflected in the design. The staff concludes that EPRI has not provided adequate requirements pertaining to the staff's position as documented in SRP Section 3.8.3 and requires that SRP Section 3.8.3 be used for considering design loading. The staff will evaluate this issue during its review of an individual application for FDA/DC.

Foundation Design for Seismic Category I Buildings

Section 4.6.1.5 of Chapter 1 requires that the design of foundations for seismic Category I buildings and structures satisfy the minimum factors of safety, taking into account the effects of seismic soil-structure interaction with respect to sliding, overturning, and floatation. The Evolutionary Requirements Document references the minimum factors of safety that are specified in SRP Section 3.8.5, "Foundations." However, as stated in the staff's letter of April 24, 1991, the effects of normal and high ground water in the design of the embedded structure and foundation were not considered. In its response of July 2, 1991, EPRI committed to include such a requirement and implemented this commitment in Revision 2 of the Evolutionary Requirements Document. The staff concludes that the requirement for foundation design for seismic Category I buildings is acceptable.

Non-Seismic Buildings and Structures

Section 4.6.1.6 of Chapter 1 requires that non-seismic buildings, structures, and structural subsystems be designed to meet the load combinations specified in the Uniform Building Code (UBC). In its letter dated April 24, 1991, the staff expressed a concern about the use of the UBC seismic Zone 2A specification that it discusses in Section 4.3.2.3 of this chapter. However, as far as load combination is concerned, the use of the UBC for non-seismic structures is acceptable.

Probability-Based Mechanistic Design Loads

Section 4.6.1.7 of Chapter 1 allows the optional use of mechanistic design loads and load combinations developed on a probabilistic basis if certain load combinations disproportionately control the design of plant structures without a rational basis. In its letter dated April 24, 1991, the staff requested that EPRI justify using the methodology in NUREG/CR-3876 "Probability Based Load Combination Criteria for Design of Concrete Containment Structures," in defining loads and load combinations without a corresponding definition of the reliability index and consistent acceptance criteria. The staff also requested that EPRI give the basis for using ANSI/ANS 51.1, 52.1, or 2.12 in defining loads and load combinations. To address these two concerns, EPRI revised the Evolutionary Requirements Document to require that probabilistic load combinations be identified and a suitable justification be prepared for NRC review and concurrence, if deemed warranted by the plant designer. Meanwhile, the rationale portion in Revision 3 of this section also was revised to acknowledge that the use of probabilistic load combinations is a departure from EPRI's commitment to the load combinations of SRP Sections 3.8.1 through 3.8.5, as indicated in Table B.1-2 of Appendix B to the Evolutionary Requirements Document, and that the plant designer is encouraged to consider their use where warranted, but regulatory concurrence will be

required on a case-by-case basis. This revision in requirements is acceptable in light of the staff-proposed position documented in the draft Commission paper issued on February 27, 1992. This position allows, in principle, the elimination of load combinations (except an SSE plus LOCA) with a probability of occurrence less than 1.0E-6 per reactor-year. However, the staff will approve such an elimination only when a specific design applicant justifies its contention with specific examples of how and where the design is governed and a comparison of structural margins with and without the load combination in question. The staff will evaluate this issue during its review of an individual application for FDA/DC.

4.6.2 Systems and Equipment

In the section entitled "Appendix A to Chapter 1" in the DSER for Chapter 1, the staff identified an open issue relative to the decoupling of the SSE from the LOCA. As discussed below and in the DSER, the staff position is that, for the design of all safety-related structures, systems, and equipment, including their supports, the loads resulting from dynamic events associated with the faulted condition (ASME Service Level D) should be combined with the LOCA loads. In its letter dated May 17, 1991, the staff requested that the rationale portion of Sections 4.6.2.3 and 4.6.2.5 and the load combinations in Tables 1.4-5 and 1.4-7 be revised to agree with the above staff position. In addition, a note should be added to Tables 1.4-5 and 1.4-7 to state that the method of combination of dynamic responses to loads is in accordance with NUREG-0484, "Methodology for Combining Dynamic Responses," Revision 1, dated May 1980. In its letter dated August 1, 1991, EPRI stated that the SSE and LOCA loads are not combined because each event is of very low probability and unrelated. The staff agrees that EPRI's position could be applicable only to piping systems in the majority of PWR plants. However, the staff's position is based on the requirements of GDC 2 of Appendix A to 10 CFR Part 50, which states that all structures, systems, and components be designed to withstand the effects of appropriate combinations of normal and accident conditions with natural phenomena. Historically, the staff has interpreted GDC 2 as requiring that the effects of the SSE and LOCA be combined for the design of all safety-related portions of the plant. Any change in this interpretation requires either an exemption from or a revision of GDC 2. Therefore, the staff position remains as stated above and as reflected in SRP Section 3.9.3. As a result, the staff will review an individual application for FDA/DC in accordance with this position.

The staff concludes that, with the exception noted above, the information in Section 4.6.2 of Chapter 1 relative to loading combinations for systems and equipment is consistent with SRP Section 3.9.3 and is, therefore, acceptable.

4.7 Design Methodology

Section 4.7 of Chapter 1 of the Evolutionary Requirements Document states that all analysis and design techniques for the ALWR will be in accordance with the industry codes and standards specified in Section 4.4 of Chapter 1. It further states that the plant designer will be required to (1) consider experience in existing LWR plants to identify design problems that have adversely affected construction costs, schedules, maintainability, or operability; (2) implement design methods and accepted advanced dynamic analysis techniques to minimize unnecessary conservatism in the plant design; and (3) develop a design approach that allows appropriate tolerance for

construction and erection problems and for potential deviations in layout and location. To implement these requirements, exceptions to several regulatory positions may be necessary. Some of these exceptions are discussed by the staff in Section 4.4 of this chapter. Others are discussed below.

4.7.1 Introduction

In its letter dated April 24, 1991, the staff requested specific examples of the accepted advanced dynamic analysis techniques referenced in Section 4.7.1.4 of Chapter 1 of the Evolutionary Requirements Document. In its letter dated July 2, 1991, EPRI noted that specific examples had been identified in Section 4.4.3 to Chapter 1 and these examples belong with the optimization subjects discussed in Appendix B to Chapter 1. The staff's evaluation of this issue can be found in Section 4.4.3 and Appendix B of this chapter.

4.7.2 Buildings and Structures

Containments

Section 4.7.2.1 of Chapter 1 requires that concrete containments be designed, constructed, and tested in accordance with ASME Code, Section III, Division 2, Subsection CC, and steel containments in accordance with ASME Code, Section III, Division 1, Subsection NE. These requirements are consistent with current LWR design practice and are acceptable.

Other Category I Buildings and Structures

Section 4.7.2.2 of Chapter 1 requires that other Category I concrete and steel buildings and structures satisfy, respectively, ANSI/ACI 349 and ANSI/AISC N690 as supplemented by the AISC load and resistance factor design method. For abnormal or extreme load combinations, the Evolutionary Requirements Document permits local yielding provided the yielding is contained and does not result in a collapse mechanism and the resulting ductility factor does not exceed that recommended in applicable regulatory positions.

In its letter dated April 24, 1991, the staff commented that the NRC has accepted the use of ductility factors for concrete and steel structures in design for impact and impulsive loadings only and not for other load combinations. In its response of July 2, 1991, EPRI acknowledged that a clarification to this requirement was necessary. Revision 3 of the Evolutionary Requirements Document states that local yielding is permissible for "impulsive and impactive" loads in the "abnormal/extreme environmental" load combinations. The staff concludes that this clarification is sufficient and that the design methodology specified for other Category I buildings and structures is acceptable.

Dynamic Analysis Techniques

Section 4.7.2.3 of Chapter 1 requires that dynamic analysis techniques comply with American Society of Civil Engineers (ASCE) 4-86, as well as other applicable codes and standards, and be qualified and proven. In its letter dated April 24, 1991, the staff commented that the NRC has not accepted all analysis techniques in ASCE 4-86. In its response of July 2, 1991, EPRI stated that ASCE 4-86 is intended to supplement the overall criteria and methodology specified in regulatory guides and SRP sections, and that

Table B.1-1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document confirms EPRI's commitment to comply with the regulatory positions except for those analysis techniques associated with the optimization subjects. This response and the requirements in this section of Chapter 1 are not acceptable because ASCE 4-86 has not been reviewed and approved by the staff. Therefore, the staff concludes that the guidelines in the SRP and regulatory guides should be used for the plant analysis and design of future ALWRs. Plant designers proposing to use ASCE 4-86 should submit a request for staff review and approval on a case-by-case basis. The staff will evaluate this issue during its review of an individual application for FDA/DC.

Timing of Seismic Analyses and Need for Confirmatory Analyses

Section 4.7.2.4 of Chapter 1 emphasizes the benefit of performing seismic analyses early in the design stage. It also requires that confirmatory analyses be performed after building structures and major equipment are in the final design stage if significant changes in the distributed mass and stiffness are introduced during the design. The staff concludes that this is a reasonable requirement and is acceptable.

Generation of Design Response Spectra or Time Histories

Section 4.7.2.5 of Chapter 1 requires that the generation of design response spectra or time histories be based on methods that minimize unnecessary conservatism. Since the Evolutionary Requirements Document did not indicate the need for NRC approval of methods such as the spectrum-to-spectrum generation procedure, the staff requested that EPRI submit a complete discussion of the limitations and verification of such procedures. In its response of July 2, 1991, EPRI did not address the NRC concern, and, therefore, the requirements in Section 4.7.2.5 of Chapter 1 of the Evolutionary Requirements Document are not acceptable. The staff position is that all analysis methods used for the licensing design basis and safety margin basis must be approved by the NRC staff. The staff will evaluate this issue during its review of an individual application for FDA/DC.

Structural Damping

Section 4.7.2.6 of Chapter 1 requires that structural damping values be based on confirmed test results, rather than conservative assumptions, whenever such data are available. In its letter dated April 24, 1991, the staff requested that EPRI submit the basis for using less conservative damping values. In its response of July 2, 1991, EPRI cited cable trays and hangers as examples for which extensive tests have been done by some utilities to generate more realistic damping values than those in RG 1.61. The staff position of using structural damping values for the design of structures is given in RG 1.61. However, the staff will not exclude the use of structural damping values based on test results. Therefore, it will evaluate the structural damping values during its review of an individual application for FDA/DC.

Masonry Walls in Category I Buildings

Section 4.7.2.7 of Chapter 1 requires that masonry walls used as temporary or permanent partitions in Category I buildings be engineered as a substructure of the building, with consideration given to the effect they could have on safety-related items, and that they be designed according to the applicable

requirements of the Uniform Building Code (UBC). The use of the UBC for the design of masonry walls in Category I structures and the use of masonry walls in Category I buildings deviate from SRP guidelines. The staff concludes that the SRP guidelines should be used for the design of masonry walls located in the Category I structures of future ALWRs. In addition, it strongly discourages the use of masonry walls in Category I structures as load-carrying members or non-load-carrying members. The staff will evaluate this issue against the criteria of Appendix A to SRP Section 3.8.4 during its review of an individual application for FDA/DC.

Concrete Expansion Anchors

Section 4.7.2.8 of Chapter 1 specifies the use of the direct-bearing or undercut type of anchor bolts to ensure the ductile behavior of the bolt when high capacity is needed and the use of wedge and sleeve anchors for small loads. In its letter dated April 24, 1991, the staff questioned the use of expansion bolts for all safety-significant applications and encouraged qualification testing under field conditions. Where expansion anchors are used, the NRC requires the use of the conservative safety factors of Inspection and Enforcement Bulletin (IEB) 79-02, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts," to account for uncertainty in field installation. In its response of July 2, 1991, EPRI acknowledged that its intent is to use expansion bolts only when necessary and that the expansion bolts will be of the undercut type (e.g., Maxibolts) in lieu of the friction type. EPRI also noted that the conservative safety factors of IEB 79-02 are intended for friction-type expansion anchors and may not apply to Maxibolts. The response is not acceptable because the issue of uncertainty in field installation was not addressed and there is no assurance that the IEB 79-02 safety factors are not applicable to Maxibolts. Therefore, plant designers should submit to the staff the safety factors they propose to use for the capacity of the Maxibolts. The staff will evaluate this issue during its review of an individual application for FDA/DC.

Stability of Shell-Type Structures Under Compression

Section 4.7.2.9 of Chapter 1 requires that the potential for global and local shell buckling be considered for shell-type structures under compression. In addition, it requires that, after appropriate consideration of the various uncertainties in materials, erection tolerances, and load description, a minimum factor of safety be maintained for all load combinations. In Revision 3 of the Evolutionary Requirements Document, the minimum factor of safety was based on ASME Code, Section III, Subsection NE, and supplemented by Code Case N-284.

Because Code Case N-284 provides lower safety factors against shell buckling than Subsection NE, the staff requested that EPRI alert the plant designer regarding the application of Code Case N-284 to asymmetric containments with large openings or provide specific conditions under which this code case can be used. In its response of July 2, 1991, EPRI stated that experienced professional containment vessel designers, working under the careful review of the utility owners and regulators, will not apply Code Case N-284 where its provisions do not apply. EPRI also stated that it was beyond the scope of the Evolutionary Requirements Document to explain the limitations of the specific code case. This is not acceptable to the staff. Subsection NE requirements

should be used for the evaluation of shell-type structures. As for Code Case N-284, the staff will evaluate its applicability during its review of an individual application for FDA/DC.

Non-Seismic Structures

Section 4.7.2.10 of Chapter 1 requires that non-seismic structures, structural subsystems, and structural components be designed to the Zone 2A specifications of the Uniform Building Code. The use of Zone 2A is discussed by the staff in Section 4.3.2.3 of this chapter.

4.7.3 Systems and Equipment

Section 4.7.3 of Chapter 1 provides discussions of design methodology requirements for systems and equipment that are more detailed than those in Section 4.4 of Chapter 1. The staff's evaluation of several of these requirements is given below.

Use of ASME Code, Section III, Appendix N

Section 4.7.3.1 of Chapter 1 states that dynamic analysis techniques for safety class components will be in accordance with Appendix N of ASME Code, Section III. Appendix N is a nonmandatory appendix that is still evolving and does not currently agree with some staff positions. Therefore, it has not been endorsed by the staff, and the staff has no immediate plans to review it. In its letter dated May 17, 1991, the staff requested that EPRI delete the reference to Appendix N and to reference applicable regulatory guides, SRP sections, or staff-approved ASME code cases in the requirement portion of Section 4.7.3.1. In its letter dated August 1, 1991, EPRI stated that only the rationale portion of Section 4.7.3.1 would be changed and that this change would only address the use of Code Case N-397. The issue of Code Case N-397 is discussed by the staff in Section 4.4.3 of this chapter. The use of Code Case N-397 is only one of several issues that are currently in Appendix N or are being proposed for future addenda to this document and that have not been endorsed by the staff. Some of these issues are damping values, use of the load coefficient method, use of the independent support motion response spectrum method of analysis, and the nonexceedance probability level in Subsection N-1725 of Appendix N. EPRI's response is not acceptable. The staff will evaluate this issue during its review of an individual application for FDA/DC in accordance with applicable SRP sections in lieu of Appendix N to ASME Code, Section III. On the basis of the above discussion, this issue is closed.

Use of ASME Code Cases N-411 and N-420 in the Same Analysis

Sections 4.7.3.2 and 4.7.3.11 of Chapter 1 allow the plant designer to use ASME Code, Section III, Code Cases N-411 and N-420 unconditionally. In its letter dated May 17, 1991, the staff requested that the following sentence be added to this requirement: "ASME Code Cases N-411 and N-420 may only be used in separate analyses and as further conditioned in RG 1.84." In its letter dated August 1, 1991, EPRI stated that since its intent to comply with RG 1.84 is indicated in Appendix B to Chapter 1 of the Evolutionary Requirements Document, this additional sentence is unnecessary. The staff's understanding of the Evolutionary Requirements Document is that a requirement in any section could override such a commitment in Appendix B. Therefore, this response is

unacceptable and the staff's position remains as stated in its letter of May 17, 1991. The staff will evaluate this issue during its review of an individual application for FDA/DC in accordance with the above position. On the basis of the above discussion, this issue is closed.

Use of Alternative Dynamic Analysis Methods

Sections 4.7.3.3 and 4.7.3.4 of Chapter 1 provide requirements that allow the plant designer to use dynamic analysis methods that have only been conditionally approved by the staff and are, therefore, not completely acceptable. These issues are discussed by the staff in Section 4.4.3 of this chapter and in the above section entitled "Use of ASME Code, Section III, Appendix N."

Requirement Applicable to Use of ASME Code Case N-411

In its letter dated May 17, 1991, the staff requested that EPRI revise Section 4.7.3.8 of Chapter 1 to clarify the use of a single damping value for both the OBE and the SSE. In its letter dated August 1, 1991, EPRI stated that the ALWR program had deleted the OBE from the design process and that the damping values in RG 1.61 for the SSE will be applicable to structures and systems except for piping, for which ASME Code Case N-411 is applicable. The elimination of the OBE is discussed by the staff in Section 4.4.3 of this chapter. The resolution of this issue may affect Section 4.7.3.8 of Chapter 1. The use of Code Case N-411, as stated in EPRI's response of August 1, 1991, is not completely acceptable. The staff requested that EPRI revise the requirement portion of Section 4.7.3.8 to include a requirement that Code Case N-411 be used only as conditioned by RG 1.84. In its response, EPRI stated that since Table B.1-2 of Appendix B to Chapter 1 indicates a commitment to comply with RG 1.84, this revision was unnecessary. The staff's understanding of the Evolutionary Requirements Document is that a requirement in any section could override such a commitment. Therefore, this portion of the response is unacceptable and the staff position remains as stated in its letter of May 17, 1991. The staff will evaluate this issue during its review of an individual application for FDA/DC, assuming that the requirement in Section 4.7.3.8 does not override the commitment to RG 1.84 in Appendix B to Chapter 1. On the basis of the above discussion, this issue is closed.

Seismic Evaluation and Design of Small-Bore Piping

Revision 3 of Section 4.7.3.13 of Chapter 1 states that seismic Category I piping 2 inches and less in nominal diameter will be analyzed on the basis of the reference spectrum approach in NCIG-14 (EPRI NP-6628), "Procedure for Seismic Evaluation and Design of Small Bore Piping," dated April 1990, unless it encompasses in-line equipment with seismic qualification requirements or is connected to sensitive component nozzles. The staff is reviewing this document as a topical report, which was submitted to the staff by the Nuclear Management and Resources Council in its letter dated March 19, 1991. Pending completion of this review, the staff's position is that the methodology in EPRI NP-6628 is not acceptable. The staff will evaluate this matter during its review of an individual application for FDA/DC.

Use of ASME Code, Section III, Subsection NF, and ANSI/AISC Standard N690

The first sentence in the requirement portion of Section 4.7.3.22 of Chapter 1 contains an acceptable commitment to the jurisdictional boundary rules of ASME

Code, Section III, Subsection NF. On the basis of EPRI's response of August 1, 1991, to a staff request for additional information dated May 17, 1991, it is the staff's understanding that the remainder of this section provides clarification only and does not permit the plant designer to take exception to this commitment. On the basis of this understanding, the staff concludes that the information in Section 4.7.3.22 is consistent with staff positions relative to jurisdictional boundary rules and is acceptable. For further clarification, the staff's position on this issue is discussed below.

The ongoing effort with regard to ASME Code, Section III referencing ANSI/AISC N690 in Subsection NF is not directly related to the issue of jurisdictional boundary. When this effort is complete and endorsed by the staff, all supports identified as falling under the jurisdiction of Subsection NF will be constructed to the rules of that subsection. Implicitly, the rules for the design portion of construction will then be in accordance with a modified version of ANSI/AISC N690.

Rules for Construction of Core Support Structures

Section 4.7.3.23 of Chapter 1 states that core support structures will be designed to the criteria in ASME Code, Section III, Subsection NG. In its letter dated May 17, 1991, the staff requested that this requirement be revised to read: "Core support structures will be constructed to the criteria specified in ASME Code, Section III, Subsection NG, where 'construction' is as defined in ASME Code, Section III, NB/NC/ND-1100(a)." In its letter dated August 1, 1991, EPRI agreed with the staff's request, except that the requirement still contains the words "designed to" rather than "constructed to," which is not completely acceptable. During its reviews of individual applications for FDA/DC, the staff will require that core support structures be constructed to the rules of ASME Code, Section III, Subsection NG, where "construction" is either as defined above or as defined in ASME Code, Section III, Subsection NG-1110. On the basis of the above discussion, this issue is closed.

Commitment to I&E Bulletin 79-02

In its letter dated May 17, 1991, the staff requested that a commitment be made in Section 4.7.3 of Chapter 1 of the Evolutionary Requirements Document that the applicable action items in I&E Bulletin 79-02, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts," Revision 2, dated November 8, 1979, will be met for pipe support base plate designs using concrete expansion anchor bolts. In its response dated August 1, 1991, EPRI stated that since the only drilled-in anchors permitted for the ALWR are to be of the "undercut" type, the safety factors in Bulletin 79-02 should not be applicable. This response is not completely acceptable. The staff's position on this issue is as follows:

- In lieu of the safety factors in I&E Bulletin 79-02, EPRI should provide the factors that will be used in the design of the undercut type of expansion anchor bolt and the basis for these factors.

- Irrespective of the type of expansion anchor bolt that will be used, the staff requires a commitment to the action item in I&E Bulletin 79-02 relative to pipe support base plate flexibility.

The staff will evaluate this issue during its review of an individual application for FDA/DC. On the basis of the above discussion, this issue is closed.

Use of ASME Fatigue Design Curves

Section 3.3 of Chapter 1 states that the plant design life for the ALWR will be 60 years. This proposed design life raises questions relative to the margins available in the current ASME fatigue design curves. These margins were established almost 30 years ago and were obtained from best-fit curves of fatigue test data by applying a factor of either 2 on stress or 20 on cycles, whichever was more conservative at each point. These factors were originally intended to cover such effects as environment, size effect, and scatter of data. However, on the basis of limited data currently available, the staff concludes that these margins may not be sufficient to account for variations in the original fatigue test data as a result of various environmental effects. In its letter dated May 17, 1991, the staff requested a commitment in Section 4.7.3 of Chapter 1 to consider such effects in the designs of applicable ASME Code Class 1 systems, components, and equipment. In its letter dated August 1, 1991, EPRI stated that if additional data or research results yield findings requiring changes to the current ASME fatigue design curves, the code consensus process will provide the proper vehicle to address such findings. The staff does not agree with all of the discussions in this response and concludes that the above commitment is not completely acceptable. The ASME Code curves may not be revised for many years. Therefore, the staff's position is that until these curves are revised, all ALWR applicants and all licensees applying for license renewal should propose appropriate fatigue design curves that will be reviewed by the staff. For the Evolutionary Requirements Document, a commitment to this position would be sufficient. Pending such a commitment, the staff will evaluate this issue during its review of an individual application for FDA/DC. On the basis of the above discussion, this issue is closed.

The staff is assessing all available data relative to environmental effects on fatigue for austenitic stainless steel and ferritic steel. The objective of this effort is to propose interim fatigue curves that may be used in lieu of the current ASME curves until ASME has acceptably addressed this concern. Pending staff implementation of these interim curves, the position stated above will be in effect.

Conclusion

The staff concludes that, with the exceptions noted above, the requirements in Section 4.7.3 of Chapter 1 of the Evolutionary Requirements Document relative to design methodology for systems and equipment are consistent with applicable SRP sections, regulatory guides, and staff positions and are acceptable.

4.8 Testing and Qualification

Section 4.8 of Chapter 1 of the Evolutionary Requirements Document provides requirements for the seismic, dynamic, and environmental qualification of mechanical and electrical equipment.

4.8.1 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

In Sections 4.4.C(7) and 4.7.A of the DSER for Chapter 1, the staff stated that EPRI had committed to revise these sections to state that the plant designer will make use of qualification by experience as permitted by governing codes and standards. This commitment was met in Revision 1 of Section 4.8.1 of Chapter 1. In addition to this commitment, Section 4.8.1 contains broad and generally acceptable requirements relative to qualification by testing, analysis, and combined testing and analysis. These qualification methods will be implemented by meeting the rules in Institute of Electrical and Electronic Engineers (IEEE) 344-1987, "Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations." This standard contains acceptable criteria with the exception of a portion of the rules relative to qualification using the seismic experience data base. In addition, Revision 2 of Table B.1-2 of Appendix B to Chapter 1 contains a commitment that EPRI will comply with RG 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants," Revision 2, which contains certain staff positions in addition to the rules of IEEE 344. However, Section 4.8.1 of Chapter 1 still contains unacceptable requirements and rationale relative to the use of the seismic experience data base. These issues are discussed below.

Seismic Qualification of Equipment by Experience

The applicable portions of the NRC's regulations governing the seismic design of nuclear power plants require that systems, structures, and components important to safety be designed to withstand the effects of earthquakes and that they be seismically qualified to perform their safety functions. The regulations stipulate that seismic qualification of such equipment will be demonstrated by either a suitable dynamic analysis or a suitable qualification test. There are no explicit provisions in the regulations that accept experience data as a means of seismic qualification.

The NRC has not approved the use of the earthquake experience data base methodology as a seismic qualification method. Rather, it has concluded that the NRC-approved, experience-based methodology, when used in conjunction with appropriate restrictions and caveats, may be an acceptable means of verifying the seismic adequacy of certain equipment in certain operating nuclear power plants.

The extent to which the NRC has approved or endorsed the application of earthquake experience data for electrical and mechanical equipment in nuclear power plants has been restricted solely to the resolution of Unresolved Safety Issue (USI) A-46, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors." In 1980, USI A-46 was formalized as a result of the safety concern that equipment needed to achieve and maintain safe shutdown in certain operating plants (i.e., those plants whose seismic licensing criteria were not reviewed for acceptability against IEEE 344-1975, RG 1.100, and SRP Section 3.10, "Seismic Qualification of Category I Instrumentation and Electrical Equipment") may not have been adequately qualified to ensure its survivability and functionality in the event of a safe shutdown earthquake. The staff discusses USI A-46 further in Section 3 of Appendix B of this chapter.

The NRC determined that it was not feasible to require those older operating plants to meet the later licensing requirements, since direct application of those criteria to older plants could have required extensive, and probably impractical, modifications of those facilities. The NRC subsequently concluded that the use of earthquake experience data with appropriate restrictions and caveats, supplemented by some test results to verify the seismic capability of equipment within certain specified earthquake motion bounds, represented the most reasonable and cost-effective means of ensuring that the purpose of the NRC regulations related to seismic design is met for those plants. One of the programmatic restrictions includes the exclusion of the application of the earthquake experience in verifying the adequacy of structures and piping.

The NRC has since received requests from utilities and industry organizations proposing the use of the earthquake experience data base methodology as a qualification method for various applications, including its application to ALWRs by EPRI. However, the NRC has not approved or endorsed any of these initiatives. In Section 4.8.1.8 of Chapter 1, EPRI proposes to use the Seismic Qualification Utility Group (SQUG) data base as one means of seismic qualification. The staff concludes that this portion of Section 4.8.1.8 is not acceptable.

Current NRC guidance (RG 1.100, Revision 2) recognizes the use of experience data as a means of seismic qualification of equipment. However, the earthquake experience data base methodology described in IEEE 344-1987 - which, as stated in RG 1.100, Revision 2, is to be evaluated by the staff on a case-by-case basis - is different from the detailed criteria and approach in the SQUG Generic Implementation Procedure (GIP). The staff does not accept the GIP as a qualification procedure. Rather, it is a verification procedure and is intended to be used only at the older operating plants under the USI A-46 resolution. Since the staff does not accept the GIP as a qualification procedure, it is not applicable to newer operating reactors or future ALWR plants. The development of the GIP verification procedures and criteria was not necessarily based on the required elements of IEEE 344-1987 or staff requirements for newer operating reactors. Thus, a significant portion of the data base in the A-46 methodology is not applicable to future ALWRs.

Therefore, consistent with RG 1.100, Revision 2, the staff will evaluate the use of experience data during its review of an individual application for FDA/DC (see Section 3 of Chapter 2 of this report for a specific application of experience data). On the basis of the above discussion, these DSER open and confirmatory issues are closed.

Number of Seismic Events Used in Equipment Qualification Programs

Section 4.8.1.1 of Chapter 1 repeats the requirement in Section 4.5.2.4.4.1 that reduces the number of full-stress cycles for 1/2 SSE from 50 to 20. This issue is discussed by the staff in Sections 4.4.3 and 4.5.2 of this chapter.

Criteria for Equipment Anchorage

In Section 4.7.A of the DSER for Chapter 1, the staff stated that EPRI had committed to revise the information in this section relative to anchorage criteria for equipment to be qualified. Section 4.8.1.3 of Chapter 1 contains a requirement that the plant designer specify the anchorage criteria for

equipment to be qualified or adopt the seismically qualified anchorage from the equipment manufacturer. The staff's interpretation of this requirement is that the plant designer will determine that the manufacturer's seismic qualification procedures are consistent with all of the ALWR qualification requirements approved by the staff. In addition, the plant designer will ensure that the "as-qualified" anchorage used in the qualification program corresponds to the "as-installed" anchorage actually used in the plant. Any differences between these two configurations will be resolved by the plant designer. The staff concludes that these requirements are consistent with current staff positions and are acceptable. Therefore, this DSER confirmatory issue is closed.

Conclusion

The staff concludes that, with the exceptions noted above, the requirements in Section 4.8.1 of Chapter 1, supplemented by the staff's position on equipment seismic qualification by experience, provide reasonable assurance that an acceptable program for seismic and dynamic qualification of electrical and mechanical equipment will be implemented for the ALWR.

4.8.2 Environmental Qualification of Mechanical and Electrical Equipment

In Table B.1-2 of Appendix B to Chapter 1 of the Evolutionary Requirements Document, EPRI has committed to comply with SRP Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment". The staff concludes that this commitment is acceptable. Section 4.8.2.1 of Chapter 1 of the Evolutionary Requirements Document states that Class 1E electrical equipment will be environmentally qualified in accordance with 10 CFR 50.49, as outlined in IEEE 323, "Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations." The staff finds that IEEE 323-1974 is acceptable as outlined in 10 CFR 50.49, but has not found IEEE 323-1983 acceptable. Where differences exist between IEEE 323 and 10 CFR 50.49, the designer must follow the NRC regulation or identify and justify the differences for the staff to review. The staff will evaluate this matter during its review of an individual application for FDA/DC.

In Table B.1-2 of Appendix B to Chapter 1, EPRI states that RG 1.89, "Qualification of Class 1E Equipment for Nuclear Power Plants," is an optimization subject. Further, Section 4.4.3.3.7 of Chapter 1 states that for environmental qualification of plant equipment, the ALWR source term will be a physically based source term in lieu of the source term specified in Atomic Energy Commission Technical Information Document (TID) 14844. As a result of reviewing the EPRI-proposed source term, the NRC staff is developing a revised source term. The staff position is that the source term should be based on the revised source term to be issued by the staff.

It should be noted that ALWRs will be designed for 60 years of operation, while the current plants are designed for 40 years. The staff concludes that the plant designer should ensure that plant equipment important to safety will be qualified for its intended service and will be able to perform its safety functions throughout its design life. The staff will address this issue during its review of an individual application for FDA/DC.

Section 4.8.2.4 of Chapter 1 states that qualification will be accomplished by physical test or by experience, demonstrating the equipment's similarity to

previously qualified equipment or to equipment that has been exposed to other more severe environments. The staff finds that the above statement can easily be misinterpreted and, therefore, needs to be clarified by stating that the method of qualification should be in accordance with 10 CFR 50.49(f). The staff will review the plant-specific designs against the requirements of 10 CFR 50.49(f), which requires that each item of electric equipment important to safety be qualified by one of the following methods:

- testing an identical item of equipment under identical conditions or under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable
- testing a similar item of equipment with a supporting analysis to show that the equipment to be qualified is acceptable
- experience with identical or similar equipment under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable
- analysis in combination with partial-type test data that support the analytical assumptions and conclusions

The staff will evaluate this matter during its review of an individual application for FDA/DC or a combined license.

Table 1.1 Design-Basis Tornado Characteristics

Region	Maximum wind speed (mph)	Rotational speed (mph)	Translational speed (mph)	Radius of maximum rotational speed (feet)	Pressure drop (psi)	Rate of pressure drop (psi/sec)
I	300	240	60	150	2.0	1.2
II	220	170	50	150	1.0	0.5
III	200	160	40	150	0.9	0.3

5 MATERIALS

5.1 Introduction

Section 5 of Chapter 1 of the Evolutionary Requirements Document provides general guidance regarding the selection of materials and the methods that will be used to fabricate nuclear power plant components. The requirements are intended to prevent degradation of the materials from a large number of causes such as intergranular attack, stress corrosion cracking, crevice corrosion, thermal and mechanical fatigue, radiation embrittlement, welding failures, bolting failures, and casting flaws.

5.2 General Requirements

5.2.1 Responsibility for Materials Selection

Section 5.2.1 of Chapter 1 of the Evolutionary Requirements Document requires that the materials for the ALWR be specified by the plant designer and have been proven suitable in service. If the plant designer proposes to use unproven material, a written justification must be provided stating why the unproven material is being selected and the basis on which it is considered to be acceptable.

The staff considers these requirements acceptable because the design choices will be constrained to ensure the chosen materials are suitable for the intended service.

5.2.2 Identification of Materials in Critical Components

Section 5.2.2 of Chapter 1 requires that the materials for critical components, the criteria for selecting the materials, and any special requirements that may be required for the selected materials be identified.

The staff considers these requirements acceptable because they will ensure that safety-related plant components will be fabricated using materials that are compatible with the intended service.

5.2.3 Codes and Standards

Section 5.2.3 of Chapter 1 requires that the specified materials meet the requirements of the applicable design and construction codes and standards and that the plant designer consider the actual service condition to determine if more restrictive conditions are needed than those imposed by code specifications. The requirements of Regulatory Guide 1.85, "Materials Code Case Acceptability - ASME Section III, Division 1," also apply to ALWRs.

The staff considers these requirements acceptable because they comply with existing NRC regulations.

5.2.4 Design-Basis Consideration

Section 5.2.4 of Chapter 1 requires that materials used in the ALWR plant meet the special conditions imposed by the design bases incorporated in the Evolutionary Requirements Document. A minimum number of types and grades of

materials must be used, where practical, consistent with service conditions and design performance objectives. The design life requirements for the materials must incorporate all required environmental and service conditions, including off-normal conditions that affect the life of the component (e.g., composition of the cooling water, steam pressure, and radiation level). Preference must be given to designs that do not push material limits and that make use of conventional materials applied well within the limits for which successful experience has been obtained. In particular, high-strength bolts or fasteners will not be used where practical; rather, sufficiently robust designs will be used that do not require special, high-strength materials.

The staff considers these requirements acceptable. The use of a minimum number of types and grades of materials will simplify plant design. Operating experience has shown that all service conditions must be considered in the selection of materials.

5.2.5 Hazardous Materials

Section 5.2.5 of Chapter 1 requires that the use of materials that present a hazard to personnel or equipment (e.g., radioactivity, toxicity, flammability, or corrosivity) be limited to those applications where no satisfactory alternatives exist. If hazardous materials must be used, the plant designer will specify shipping, storage, handling, and usage requirements for the material to minimize the hazards and satisfy applicable regulations and practice. A building, meeting the requirements of the Environmental Protection Agency, will be provided for storage of hazardous and toxic wastes awaiting disposal.

The staff concludes that these requirements do not conflict with the Commission's regulations are, therefore, acceptable.

5.2.6 Review of LWR Experience

Section 5.2.6 of Chapter 1 requires that plant designers, when selecting ALWR materials, consider the lessons learned from the operation of existing plants and the measures necessary to prevent problems from recurring. The ALWR plant designer will conduct a specific review of LWR experience (both domestic and foreign) to identify significant materials problems and proven methods of resolving them. This review will supplement and update the review of experience that is inherent in the Evolutionary Requirements Document. For the review, the designer should fully use information available at the time the plant is designed from such sources as EPRI-sponsored research and studies and from government, academic, and industry sources. The problems identified in this review will be listed in a report to be submitted to the plant owner, and appropriate materials requirements will be incorporated into the design of the ALWR plant.

The staff considers these requirements acceptable. The incorporation of lessons learned from operation of existing plants will ensure that materials problems experienced in these plants will be prevented from recurring.

5.2.7 Metallic Materials

Section 5.2.7 of Chapter 1 requires that metallic materials in contact with reactor coolant be corrosion resistant, such as austenitic stainless steel or

carbon and low-alloy steels, with an adequate consideration of corrosion; resistant to detrimental forms of corrosion, such as intergranular attack, stress corrosion cracking, or contact corrosion between different materials; and restricted in cobalt content to as low a level as practical for all components that are made of stainless steel or nickel-based alloy and that have a large wetted surface area (e.g., steam generator tubing, major piping, cladding). For such components fabricated with stainless steel or nickel-based alloys, the cobalt content will be restricted to 0.020 weight percent or less. Cobalt-based alloys will be avoided except in cases for which no proven alternative exists. The plant designer will identify to the plant owner all applications of cobalt-based alloys in wear applications and state the basis for their use. In addition, ferritic pressure boundary materials will be resistant to brittle fracture and will satisfy Appendix G to 10 CFR Part 50. The nil ductility temperature (RT_{NDT}) of these materials will not exceed 10 °F.

Lead, antimony, cadmium, indium, mercury, zinc, bismuth, and tin metals and their alloys will not be allowed to come in contact with reactor coolant system (RCS) primary components or secondary components made of stainless steel or Inconel. For bearings in the secondary system, the plant designer will demonstrate that the bearing design is such that harmful amounts of material with a low melting point will not enter the feedwater to jeopardize stainless steel or Inconel components.

Zinc may be added to the coolant in BWRs in limited, controlled amounts. Copper alloys may be used for very limited, controlled applications in BWRs, such as the low-temperature pump bearings, where the material cannot enter the reactor coolant.

EPRI states that the plant designer will select materials for the reactor vessel support structures that are resistant to brittle fracture and experience a minimal shift in RT_{NDT} as a result of neutron fluence for the support scheme used. The plant designer will provide an analysis of the reactor vessel support structure to demonstrate its acceptability for the maximum design plant life and capacity factor and to identify any operational constraints to avoid the potential for brittle fracture. Sufficient access will be provided to the reactor vessel support structure to enable monitoring of the material temperature and performing modifications or heat treatment of the structural materials, if required.

The staff considers these general requirements acceptable because only corrosion-resistant materials will be used in contact with the reactor coolant. Also, ferritic materials are required to satisfy the requirements of Appendix G to 10 CFR Part 50. Metals and alloys with low melting points will not be used in contact with the reactor coolant.

To reduce general radiation fields resulting from the presence of cobalt-60 in the oxide layer of the RCS piping, zinc may be added to the coolant in BWRs in limited, controlled amounts. Zinc injection reduces the radiation fields by replacing the cobalt with zinc in the piping oxide layer. One of the side effects of zinc injection is the creation of zinc-65, which increases piping dose rates and requires special consideration during radioactive waste disposal. EPRI is investigating a way to solve this problem by using a zinc

isotope depleted in zinc-64. The staff will review this issue again at the vendor application stage to determine what advances have been made in this area.

The staff finds the restrictions on the use of cobalt materials in contact with reactor coolant to be acceptable because activated cobalt-60 is a major source of occupational exposure in nuclear power plants.

5.2.8 Non-Metallic Materials

Section 5.2.8 of Chapter 1 requires that the impurity levels of non-metallic materials used within the nuclear steam supply system (NSSS) and associated systems be controlled to the limits specified below. These limits apply to specific locations within the plant.

For PWR RCS applications, non-metallic materials will meet the impurity limits specified below under any of the following conditions:

- The material is in contact with reactor coolant in service.
- The material is applied to surfaces that will be in contact with reactor coolant and the material is not completely removed before service.
- The material is in contact with stainless steel or nickel alloys on external surfaces of RCS components either by design or because it is not completely removed before service.
- The material is exposed to a radiation dose greater than 1E+5 rads during component life.

	Impurity Limits
Chlorine	200 parts per million (ppm) maximum
Fluorine	200 ppm maximum
Mercury (Hg)	1 ppm maximum
Arsenic (As)	2 ppm maximum
Lead (Pb)	10 ppm maximum
Sulfur (S)	200 ppm maximum
Zinc (Zn)	200 ppm maximum
Combined Hg, As, Pb, S, Zn	300 ppm maximum

Examples of unacceptable non-metallic materials are polyvinylchloride (PVC), polytetrafluoroethylene (Teflon), fluorosilicones, and neoprene.

For PWR secondary system components where the non-metallic materials may contaminate the feedwater or are in the form of substances applied to clean stainless steel or nickel-based alloys used in secondary system components, the impurity limits are as specified below:

Total chlorine plus fluorine	500 ppm maximum
Heavy metals (total Hg, As, Pb)	250 ppm maximum
Total sulfur	500 ppm maximum

For the balance of PWR applications, non-metallic materials used inside the containment that can come in contact with steel or nickel-based alloys during fabrication, shipping, and storage will not contain more than the following:

Total chlorine	500 ppm maximum
Total fluorine	300 ppm maximum
Active sulfur	700 ppm maximum
Elements in metallic form with a low melting point	500 ppm maximum total or 200 ppm maximum for any individual element

For BWR applications, non-metallic materials that are in contact with reactor coolant during plant operation, either as a result of design or because they are not completely removed after fabrication or installation, and that remain in direct contact with stainless steel or nickel-based alloys will meet the limits specified above for the balance of PWR applications.

The staff considers these requirements acceptable because non-metallic materials will be controlled to acceptable impurity levels. This will ensure that non-metallic materials do not adversely affect the corrosion resistance and ductility of metal components. However, EPRI should revise the Evolutionary Requirements Document to include limits on nitrites, nitrates, and total halogens as chlorine. In addition, a total limit on total chlorine + total sulfur + total nitrite + total nitrate expressed as mole-equivalents of chlorine should also be included. Pending such a revision, the staff will evaluate an individual application for FDA/DC or a combined license to ensure that such limits are imposed.

5.3 Materials Selection

5.3.1 Materials in the Reactor Coolant System and Related Systems

Wrought Austenitic Stainless Steels

Section 5.3.1.1 of Chapter 1 of the Evolutionary Requirements Document specifies that the use of austenitic stainless steel will be subject to the following requirements:

- Materials will be in the solution heat-treated condition, or solution heat treated at a later step, regardless of service temperature. Heat treatment will be done at 1900 °F minimum metal temperature, followed by a qualified cooling process.

- Grain size and uniformity will be controlled in the material to ensure that adequate ultrasonic tests (UTs) can be performed where required. Requirements for attenuation may be used instead of controlling grain size and uniformity.

- Materials for use at temperatures above 200 °F in borated water or that are part of the RCS pressure boundary will be tested to verify freedom from sensitization.

- Cold-work control of materials for service above 200 °F in borated water or that are part of the RCS pressure boundary will meet the following requirements:

  - Hardness of austenitic stainless steel raw materials will not exceed 92 HRB.

  - Hardness will be controlled during fabrication by process control of bending, cold forming, straightening, or other similar operation.

- In PWRs, cold-worked austenitic stainless steel may be used for small parts (e.g., pins, fasteners, and sleeves) if no proven alternative is available; however, in those cases, the following specific requirements will be met:

  - All such applications of cold-worked stainless steel will be identified as "critical."

  - The technical justification of each application will be documented, including relevant laboratory and service experience.

  - The technical justification will include a description of the process controls that will be applied to ensure the material is subjected to the proper amount of cold work.

- The following manual grinding controls are applicable to a BWR and are desirable but not required for a PWR:

  - Grinding performed before solution heat treatment requires no additional control except that after heavy grinding, light grinding or flapping will be required.

  - Grinding will be restricted except for fit-up, nondestructive testing, or to remove scratches or surface defects.

  - All grinding of austenitic stainless steel surfaces exposed to reactor water in service that are not subsequently solution heat treated will be performed in accordance with a written procedure. The procedure will involve finishing the ground surface with successively finer grit sizes to remove the bulk of cold-worked material, with the final grit size no coarser than #120 grit.

- For all stainless steel welding materials including consumable inserts for components that operate above 200 °F, in borated water, or that are part of the RCS pressure boundary, the average ferrite content will be in the range of 5 FN (ferrite number) to 13 FN (ferrite content of undiluted weld deposits will be determined by magnetic measurements as specified in ASME Code, Section III, Division 1 (Winter 1976 or later).

The staff's evaluation regarding intergranular stress corrosion cracking of austenitic stainless steel is provided in Section 5.3.1.8 of this chapter.

In addition, the staff requires that the grinding controls also be applied to PWR applications. The staff will evaluate this issue during its review of an application for a combined license.

The staff concludes that, with the exception noted above, the implementation of these requirements will ensure that wrought austenitic stainless steel will perform in service as designed.

Martensitic Stainless Steel

Section 5.3.1.2 of Chapter 1 requires that the use of wrought and cast martensitic stainless steel be subject to the following requirements:

- All components made from wrought and cast forms of 12 percent chromium martensitic stainless steel will be properly heat treated for the intended service. Proper heat treatment consists of normalized and tempered or quenched and tempered. For applications in which stress corrosion cracking has been identified as a concern, the heat treatment will be controlled to limit the hardness. The hardness limits can vary depending on application. Hardness limits will be specified on the basis of existing industry experience and on relevant ASME codes and ASTM standards.

- Weld repairs on all wrought and cast parts will be followed by a full heat treatment. Localized post-weld heat treatment is not permitted.

In Section 5.3.A.2 of the DSER for Chapter 1, the staff determined that the ALWR requirements regarding hardness of martensitic and precipitation-hardened stainless steel were insufficient. The hardness of martensitic and precipitation-hardened stainless steel must be kept below that specified in the nuclear industry today. The hardness of 40 (equivalent Rockwell scale of C) is too high for any application; a hardness of 25 may even be too high for some applications. The harder the stainless steel, the lower the fracture toughness and the greater its susceptibility to stress corrosion cracking.

EPRI revised Sections 5.3.1.2.1 and 5.3.1.6.6 of Chapter 1 to state: "Hardness limits shall be specified on the basis of existing industry experience and on relevant ASME codes and ASTM standards." Therefore, an individual FDA/DC or combined license applicant will be required to specify hardness limits on the basis of Section III of the ASME Code and American Society for Testing and Materials (ASTM) standards. The staff will evaluate this matter during its review of an individual application for FDA/DC or a combined license. On the basis of the above discussion, this DSER confirmatory issue is closed.

The staff concludes that the implementation of these requirements will ensure that martensitic stainless steels will perform in service as designed.

Nickel, Chromium, and Iron (Ni-Cr-Fe) Alloys

Section 5.3.1.3 of Chapter 1 requires for the PWR that Alloy 600 be restricted to applications outside the pressurizer and to applications for which low-carbon stainless steel cannot be used. For PWR applications requiring the use of the special properties of Alloy 600 (e.g., resistance to chloride stress corrosion cracking and low thermal expansion coefficient), the application

must be justified. Justification should be based on need, the lack of a suitable alternative, evidence of successful service performance under the specific conditions anticipated, and a review of relevant laboratory experience. Ease of inspection and replacement also will be considered. Alloy 600 will be given a special thermal treatment to improve resistance to stress corrosion cracking. The use of Alloy 690 will be restricted to steam generator tube applications.

For BWR applications requiring the use of the special properties of Ni-Cr-Fe alloys (e.g., strength or low thermal expansion coefficient), the applications must be justified. Justification should be based on need, the lack of a suitable alternative, evidence of successful service performance under the specific conditions anticipated, and a review of relevant laboratory experience. Ease of inspection and replacement also will be considered.

Pickling of wetted surfaces for all Ni-Cr-Fe alloys is prohibited for both BWRs and PWRs.

In Section 5.3.A.3.b of the DSER for Chapter 1, the staff identified a concern regarding the use of Alloy 600 in future ALWRs. Section 5.3.1.3.1 of Chapter 1 of the Evolutionary Requirements Document specifies that Alloy 600 can be used in ALWRs. EPRI has revised Section 5.3.1.3 to place restrictions on the use of Alloy 600. EPRI states that the designer will not use Alloy 600 in the steam generators and pressurizers. For PWR applications, Alloy 600 will be given a special thermal treatment to improve resistance to stress corrosion cracking. However, in general, the staff considers Alloy 600 undesirable for use in the ALWR because of the stress corrosion cracking experienced in existing nuclear plants. It specifically discourages the use of Alloy 600 in steam generators and pressurizers. Therefore, it will require that the applicant for any standard design application identify the use of Alloy 600 and provide information concerning its use. Those applications will be reviewed and approved by the staff on a case-by-case basis. In addition, the use of other Ni-Cr-Fe alloys such as Alloy 690 or 800 should be considered in applications for which primary water stress corrosion cracking is a concern. These applications also will be reviewed on a case-by-case basis. On the basis of the above discussion, this DSER confirmatory issue is closed.

The staff concludes that, with the exception noted above, the implementation of these requirements will ensure that Ni-Cr-Fe alloys will perform in service as designed.

Austenitic Stainless Steel Castings

Section 5.3.1.4 of Chapter 1 requires that austenitic stainless steel castings meet the following requirements:

- All castings will be solution heat treated after casting to a minimum of 1950 °F metal temperature followed by a qualified cooling process.

- For all austenitic stainless steel castings that operate in a water or steam environment at a temperature of 200 °F or greater, in borated water or that are part of the RCS pressure boundary, the ferrite content will be between 8 FN and 30 FN. The ferrite content will be determined using ASTM A800.

- Radiographic quality will be controlled by the applicable codes and standards.

- The filler metal used for weld repairs will be 308L, and the average ferrite content of the filler metal will be a minimum of 5 FN and a maximum of 13 FN. Measurements will be made on the as-deposited undiluted weld pads. For hard surfacing applications, loss of delta ferrite during subsequent solution heat treatment will be limited to an acceptable level that will be demonstrated by process qualification.

- To minimize the detrimental effects of thermal aging in cast austenitic stainless steels, the following metallurgical factors will be controlled:

  - Ferrite content will not exceed 30 FN.

  - Molybdenum will not be used as an alloying element unless it can be demonstrated that sigma phase embrittlement will not affect design requirements.

  - Analysis and/or accelerated testing will be performed to ascertain if the toughness will not decrease below the limit required by design at the end of the component service life.

  - Weld joints between austenitic stainless steel castings will not be used if inservice ultrasonic inspection of the weld between the castings will be required.

The staff concludes that the implementation of these requirements will ensure that austenitic stainless steel castings will perform in service as designed.

Carbon and Low-Alloy Steel Materials

Section 5.3.1.5.1 of Chapter 1 requires that allowance be made for corrosion (general, pitting, and crevice) of any unprotected carbon steel materials exposed to a water environment. This allowance will be based on industry experience at the time the plant is designed. The review of experience will specifically include establishing a corrosion allowance and identifying its technical basis. In addition, the following requirements will be met for the material product forms listed below.

Pressure Vessel Steel

Section 5.3.1.5.2 of Chapter 1 specifies the following requirements for pressure vessel steel:

- SA533 Grade B Class 1 plate and SA508 Class 1, 2, and 3 forging materials will be used for primary coolant pressure boundary components.

- PWR pressure vessels and piping made of low-alloy or carbon steel in contact with the reactor coolant will be clad with austenitic stainless steel. The cladding will have a minimum of deposited ferrite between 5 FN and 13 FN.

- For BWRs, unclad low-alloy or carbon steel may be used for such items as the reactor vessel head, nozzle areas, and piping systems. An appropriate allowance for corrosion will then be used, and the cleanup system will be sized appropriately to remove resultant corrosion products. If hydrogen water chemistry is used, the reactor vessel will be completely clad with austenitic stainless steel. In this case, technical justification of the type of cladding material and the cladding process must be provided on the basis of past experience and the specific environmental conditions.

Seamless Pipe

Section 5.3.1.5.3 of Chapter 1 specifies that seamless plain carbon steel pipe be SA333-6. Wherever possible, seamless pipe is to be used instead of welded pipe for Class 1 systems inside the containment. The SA333-6 piping will be normalized or normalized and tempered.

Welded Pipe

Section 5.3.1.5.4 of Chapter 1 requires that welded plain carbon steel pipe be SA671-Grade CC70 or SA333-6. The impact properties of the finished pipe and weld metal will be 13 ft-lb minimum at -50 °F. The welded pipe will be normalized or normalized and tempered.

Plate

Section 5.3.1.5.5 of Chapter 1 requires that plain carbon steel plate be SA516. The impact properties will be 13 ft-lb minimum at -50 °F.

Forgings

Section 5.3.1.5.6 of Chapter 1 requires that plain carbon steel forgings be SA350-Grade LF2, SA508 Class 1, or SA105. The forgings will be heat treated by normalizing or normalizing and tempering. Impact properties will meet the requirements for SA350-Grade LF2.

Castings

Section 5.3.1.5.7 of Chapter 1 requires that plain carbon steel castings be SA352-Grade LCB or SA216 WCB. Impact requirements for SA352-Grade LCB will be met. Cast iron will not be used in essential functional parts of safety-related components such as valve yokes.

Fittings

Section 5.3.1.5.8 of Chapter 1 requires that plain carbon steel fittings be SA420-Grade WPL-6.

Evaluation

In Section 5.3.A.5 of the DSER for Chapter 1, the staff stated that the corrosion allowance for the carbon and low-alloy steel in the Evolutionary Requirements Document was inadequate because experience at operating nuclear plants has shown that the accepted standard corrosion allowance

is inadequate. The plant designer should consider all types of corrosion (i.e., microbiological, pitting, and crevice) in the design. EPRI added the following to Section 5.3.1.5 of Chapter 1: "The allowance shall be in conformance with current industry experience." The staff concludes that the above revision is unacceptable because conformance with current industry experience lacks specificity.

The staff concludes that the plant designer should control corrosion mechanisms, such as general, pitting, crevice, and microbiological corrosion, in the design of piping systems of primary and secondary systems. The general corrosion allowance should comply with the corrosion allowance specified in Section III of the ASME Code and ANSI/ASME B.31.1, "Power Piping." For the specific corrosion allowance (such as that for microbiological corrosion), the plant designer may use industry methodologies such as the EPRI computer code CHECMATE. However, the staff will have to review and approve any industry methodology before it is used in the ALWR design. The staff will evaluate this matter during its review of an individual application for FDA/DC. On the basis of the above discussion, this DSER confirmatory issue is closed.

The staff concludes that, with the exception noted above, the implementation of the requirements in Section 5.3.1.5 of Chapter 1 will ensure that carbon and low-alloy materials perform in service as designed. However, the impact properties of welded pipe and plate material also must meet the requirements of Table NB-2332(a)-1 of Section III of the ASME Code.

Precipitation-Hardened Stainless Steel

Section 5.3.1.6 of Chapter 1 requires that precipitation-hardened stainless steel meet the following requirements:

- Precipitation-hardened stainless steel will be in accordance with the ASME requirements for SA564 Type 630 (17-4PH). No other precipitation-hardened stainless steel is permitted unless specifically identified in the intended application and justified on the basis of need, lack of suitable alternatives, and good service experience in the specific application being considered.

- The material is to be used in the solution heat-treated and aged condition. The material will be heat treated to provide the required mechanical properties and resistance to stress corrosion cracking. The minimum aging temperature will be 1075 °F.

- The material will not be used if irradiation or elevated temperature will cause the material to be unsuitable for service.

- Welding is not permitted after final heat treatment.

- Any forming or bending of parts will be done before the aging heat treatment.

- Maximum hardness limits will be specified on the basis of existing industry experience and relevant ASME Codes and ASTM Standards.

The staff concludes that the implementation of these requirements will ensure that precipitation-hardened stainless steel will perform in service as designed.

Ni-Cr-Fe Alloy X-750

Section 5.3.1.7 of Chapter 1 specifies that Ni-CR-Fe Alloy X-750 will be used in the ALWR only in those applications where a lower strength material is impractical and for which a substantial base of successful experience exists. Where Inconel Alloy X-750 is used, the application will be in accordance with EPRI NP-6202, "Material Specification for Alloy X-750 in LWR Internal Components."

The staff considers these requirements acceptable because improperly heat-treated Alloy X-750 is susceptible to cracking and EPRI NP-6202 provides the technical basis for the use of this alloy.

Prevention of Intergranular Stress Corrosion Cracking of Austenitic Stainless Steel

Section 5.3.1.8 of Chapter 1 specifies that austenitic stainless steel must be resistant to intergranular stress corrosion cracking (IGSCC). All austenitic stainless steel that is in contact with BWR reactor coolant at a temperature above 200 °F during power operation or that is part of the BWR RCS pressure boundary or that is in borated water at any temperature in a PWR and that is welded without subsequent solution heat treatment will meet the following requirements:

- For PWR service, only low-carbon wrought austenitic stainless steel, which includes Types 304L, 316L, 304NG, 316NG, and modified 347, will be used.
- For BWR service, only low-carbon wrought austenitic stainless steel, Types 304NG, 316NG, and modified 347 with a maximum carbon content of 0.020 percent, will be used.
- These materials will be tested for resistance to sensitization in accordance with Section 5.4.2.4 of Chapter 1 of the Evolutionary Requirements Document.
- Incoming components, parts, raw materials, and heat-treated parts will be examined for excessive intergranular attack (IGA) unless a minimum of 0.030 inch of metal is removed from all as-received surfaces during fabrication.

In Section 5.3.A.1 of the DSER for Chapter 1, the staff stated that EPRI should revise the material selection for reactor coolant pressure boundary piping to prevent IGSCC of austenitic stainless steels. The staff has recommended that licensees and applicants follow Revision 2 of NUREG-0313 "Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping," to prevent IGSCC in stainless steel. EPRI has revised Section 5.3.1.8 of Chapter 1 of the Evolutionary Requirements Document

to reference this document. In addition, in its letter dated February 3, 1992, EPRI revised Table B.1-2 of Appendix B to Chapter 1 to require the use of Revision 2 of NUREG-0313. The staff concludes that these revisions are acceptable, and, therefore, this DSER open issue is closed.

In addition, it is important that adequate field and shop fabrication processes be used to minimize the sensitization of materials to IGSCC. Therefore, the staff will evaluate this matter during its review of an individual application for a combined license.

The staff concludes that, with the exception noted above, the implementation of the requirements in Section 5.3.1.8 of Chapter 1 will ensure that austenitic stainless steel will be resistant to IGSCC in service.

5.3.2 Materials in Feedwater, Steam, and Condensate Systems

Section 5.3.2 of Chapter 1 imposes the following requirements on materials in feedwater, steam, and condensate systems:

- The use of copper alloys is prohibited for PWR components that will be in contact with feedwater, steam, or condensate. Copper alloys may be used in BWRs in certain applications for service conditions of 200 °F or less; however, all such applications will be identified by the plant designer, who will document the basis for their acceptability on the basis of current experience.

- The ALWR design will incorporate into the secondary system component design the material considerations identified in EPRI NP-2294, "Guide to Design of Secondary Systems and Their Components To Minimize Oxygen-Induced Corrosion."

- Corrosion/erosion-resistant materials will be used for all components exposed to wet steam or flashing liquid flow where significant erosion could occur. The degree of corrosion/erosion resistance of the material will be consistent with the temperature, moisture content, and velocity of the wet steam to which the component is exposed. Plain carbon steel with no deliberate alloying additions other than carbon and manganese will be used for this application.

The staff concludes that the implementation of these requirements will ensure that materials in feedwater, steam, and condensate systems will perform as designed.

5.3.3 Fasteners and Adhesives

Section 5.3.3 of Chapter 1 imposes the following requirements on metallic fasteners and adhesives:

- The materials for threaded fasteners used to maintain pressure boundary integrity in the reactor coolant and related systems and in the steam, feedwater, and condensate systems and the threaded fasteners used inside those systems and in pipe and component structural mountings for those

systems will be selected and specified by the plant designer on the basis of their previous satisfactory performance in similar applications. Similarity will be based on

- comparable temperature and environment, including radiation dose and abnormal conditions such as wetting by gasket leakage
- comparable stresses, including primary, secondary, and peak stresses, and comparable design details that may affect the stresses (e.g., thread form, head configuration, fits, and tolerances)
- comparable service cycles, including magnitude and frequency
- comparable fabrication and installation (e.g., heat treating, plating or other surface treatments, thread-forming, head-forming, cleaning, lubricating, and preloading)
- comparable inspection during fabrication and installation and while in service

- The application of threaded fasteners will be in accordance with the requirements of Section 12.7 of Chapter 1 of the Evolutionary Requirements Document and with the guidelines of EPRI NP-6316, "Guidelines for Threaded-Fastener Application in Nuclear Power Plants."

- The lubricants to be used on all threaded fasteners that will maintain pressure boundary integrity in the reactor coolant and related systems and in the steam, feedwater, and condensate systems and the threaded fasteners used inside those systems and in pipe and component structural support for those systems will be completely specified by the plant designer in appropriate drawings and specifications. That is, field selection of thread lubricants will not be permitted. The thread lubricants will be selected on the basis of experience and test data that show they are effective, but will not cause or accelerate corrosion of the fastener. If leak sealants are used on threaded fasteners or can be in contact with the fastener in service, their selection will be based on satisfactory experience or test data. The plant designer will consider possible adverse interaction between sealants and lubricants.

- Acceptable non-metallic adhesives are

  - silicone compounds for continuous service below 400 °F
  - polyether urethanes for continuous service below 200 °F

The staff concludes that the implementation of these requirements will ensure that metallic fasteners and adhesives will perform in service as designed.

5.3.4 Thermal Insulation Materials

Section 5.3.4 of Chapter 1 imposes the following requirements on thermal insulation materials:

- Metallic insulation (Type 304) or blanket insulation with metallic jackets will be required for piping and components where inservice inspection or possible contamination make non-metallic materials unsuitable.

- For the use of non-metallic insulation of austenitic stainless steel materials, the guidance of Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel," should be followed.

The staff concludes that these requirements are acceptable because metallic insulation has been proven to perform satisfactorily in service. Non-metallic insulation will be used in accordance with the guidance in Regulatory Guide 1.36.

5.3.5 Seals, Gaskets, Packing, Sealants, Paints and Protective Coatings, Lubricants and Hydraulic Fluids, and Cleaning, Packaging, and Storage Materials

Section 5.3.5 of Chapter 1 requires that the plant designer conduct a program for evaluating the effects of intended products on other ALWR components under normal and postaccident conditions. For each product evaluated, the designer will provide recommendations and limits for its use in the steam supply systems and other applications in the ALWR. The designer will rely on information from component vendors only when substantiated by operational experience.

The staff concludes that the implementation of these requirements will ensure that seals, gaskets, packing, sealants, paints, and protective coatings, lubricants and hydraulic fluids, and cleaning, packaging, and storage materials are selected on the basis of operational experience. However, the staff will require that the FDA/DC applicant specify the specific ANSI standard covering this subject. The staff will evaluate these components, as appropriate, during its review of an individual application for FDA/DC.

5.3.6 Electrical Materials

Section 5.3.6 of Chapter 1 requires that the plant designer review materials used in the plant's electrical systems, particularly those used in safety-related applications, for functional reliability during normal, abnormal, plant operation, and accident conditions. The fire-retardant characteristics of the materials used in the electrical systems will be addressed to minimize the probability of fire and the consequences should a fire occur.

The staff considers this requirement acceptable. However, it is not clear if it is sufficient to address such issues as aging of cable insulation and other electrical materials over the design life and full range of environmental conditions. The staff will evaluate this issue during its review of an individual application for FDA/DC.

5.3.7 Weld Materials

Section 5.3.7 of Chapter 1 imposes the following requirements on welding materials:

- The strength and toughness of the ferritic steel welds will be equivalent to that of the base metal.

- The welding consumables will meet the requirements of the ASME (SFA) or the American Welding Society specification as appropriate for code or non-code construction.

These requirements are in accordance with Section III of the ASME Codes and are, therefore, acceptable.

5.4 Process Controls

5.4.1 Surface Condition

Section 5.4.1 of Chapter 1 of the Evolutionary Requirements Document requires that fabrication and installation processes adversely affecting the surface condition or microstructure (e.g., forming, bending, welding, heat treating, and surface grinding) be controlled to ensure the product meets the engineering requirements. Cleanliness standards during fabrication and subsequent handling and storage will be adopted on the basis of practice and standards that are current when the ALWR is fabricated.

The staff concludes that this requirement is acceptable because installation processes will be controlled through all stages, thus ensuring trouble-free operation.

5.4.2 Fabrication Controls

Section 5.4.2 of Chapter 1 imposes requirements related to the fabrication and welding of materials for general ALWR applications. As a minimum, this control will be applied to the fabrication steps given below for the materials and processes involved.

Ferritic Steels

Preheat and post-weld heat treatment of ferritic steels will be controlled in accordance with NRC Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel." Applications in areas of limited accessibility will be qualified in accordance with Regulatory Guide 1.71, "Welder Qualifications for Areas of Limited Accessibility." Heat inputs in cladding will be in accordance with Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components."

Austenitic Stainless Steels

Austenitic stainless steel will be fabricated in accordance with Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." Interpass temperatures will be controlled to improve resistance to IGSCC. Stainless steels will be welded in accordance with Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Material," and applications in areas of limited accessibility will be qualified in accordance with Regulatory Guide 1.71.

Martensitic Stainless Steels

Martensitic stainless steels will be heat treated followed by proper tempering to prevent stress corrosion cracking (SCC) and hydrogen embrittlement.

Precipitation-Hardenable Stainless Steels

Precipitation-hardenable stainless steels will be heat treated with age hardening and tempering to prevent SCC and hydrogen embrittlement.

Nickel-Based Alloys

Nickel-based alloys will be properly heat treated, and surface contamination will be controlled to prevent intergranular penetration.

Alloys in General

Copper penetration into base metal and detrimental chromium carbide precipitation will be eliminated to prevent IGSCC. Surface peening also will be controlled in PWR applications to prevent cracking. Deposition control welding or heat welding will be used to modify residual stresses in accordance with the guidelines of NUREG-0313, Revision 2.

Conclusion

The staff concludes that these requirements are acceptable because they comply with the guidelines of NUREG-0313.

5.4.3 Examination and Tests

Section 5.4.3 of Chapter 1 requires that material and components be tested during and following fabrication for compliance with service requirements. As a minimum:

- Pressure-retaining material will be tested for mechanical properties and chemical composition to ensure conformance with the material specification.

- Pressure-retaining material, including weldments, will be examined by code-required nondestructive methods including ultrasonic, radiographic, and magnetic or liquid penetrant examination.

- Material used for tensile and impact test specimens will be heat treated in accordance with the appropriate code.

- For PWR applications of austenitic stainless steel materials, the ASTM A 708 Strauss Test or the ASTM A262 Practice E, Modified Strauss Test, will be used to demonstrate freedom from sensitization in fabricated, unstabilized stainless steel. For austenitic stainless steel materials in BWR applications, the ASTM A262, Modified Practice A, or the electrochemical potentiokinetic reactivation test will be used.

- For joints that are not examined volumetrically and for which access is limited to 14 inches or less in two directions, the welder qualification procedure will provide for testing of the welder under simulated access conditions.

The staff concludes that these requirements are acceptable because they comply with Section III of the ASME Code and Regulatory Guides 1.44, and 1.71.

5.4.4 Shipping and Storage

Section 5.4.4 of Chapter 1 requires that all materials and components be suitably protected from damage as a result of environmental conditions in accordance with the requirements of ANSI N45.2.2, "Packaging, Shipping, Receiving, Storage, and Handling of Items for Nuclear Power Plants."

The staff concludes that this requirement is acceptable because it complies with ANSI N45.2.2 as endorsed by the staff.

5.4.5 Installation

Section 5.4.5 of Chapter 1 requires that all materials and components be handled during plant construction in accordance with the housekeeping requirements of ANSI N45.2.3, "Housekeeping During the Construction Phase of Nuclear Power Plants."

The staff concludes that this requirement is acceptable because it complies with ANSI N45.2.3 as endorsed by the staff.

5.4.6 Flush, Hydro, and Layup

Section 5.4.6 of Chapter 1 requires that the requirements and recommendations of ANSI N45.2.1, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants," apply for onsite cleaning of materials and components, cleanliness control, and preoperational cleaning and layup of water-cooled nuclear power plant fluid systems.

The staff concludes that this requirement is acceptable because it complies with ANSI/N45.2.1 as endorsed by the staff.

5.5 Environmental Conditions

5.5.1 Range of Environmental Conditions

Section 5.5.1 of Chapter 1 of the Evolutionary Requirements Document requires that materials selected for use in the ALWR be compatible with the full range of environmental conditions that may be encountered over the plant life. These environmental conditions include temperature, humidity, radiation, chemistry of fluids or materials in contact with the material, and other external conditions that may affect the suitability of a material. The plant designer will document the environmental conditions used as the basis for selecting ALWR materials. These environmental conditions will be consistent with the specific requirements in other chapters of the Evolutionary Requirements Document.

The staff concludes that these requirements are acceptable because the selected materials will be compatible with the full range of environmental conditions throughout the plant life. In addition, the use of radiation-damage-resistant materials in high-radiation areas will maximize their service life, reduce the frequency of replacement, and thereby, reduce personnel radiation exposure.

5.5.2 Water Chemistry Design Basis

BWR Water Chemistry Design Basis

Section 5.5.2.1 of Chapter 1 requires that the water chemistry design basis for BWR plant systems be in accordance with EPRI NP-4947-SR, "BWR Hydrogen Water Chemistry (HWC) Guidelines," 1987 Revision, and its subsequent revisions, and as supplemented by the guidelines in Table 1.2 in this chapter. The specific HWC control values in EPRI's guidelines relative to recirculating piping (e.g., 230 millivolts) will apply to nozzles, components, and other nonreplaceable components in the reactor vessel lower plenum.

EPRI addresses the use of HWC for the advanced BWR design. However, the use of HWC at plants such as Hatch, Brunswick, and Duane Arnold has resulted in unexpectedly high operational and post-shutdown radiation levels in reactor coolant system piping. EPRI has acknowledged the potential drawbacks of using HWC and has stated that investigations are under way to identify a solution to some of the problems resulting from the use of HWC. A special evaluation will be made when considering carbon and low-alloy material for reactor coolant service with less than 10 ppm oxygen as a result of HWC. The evaluation will include erosion/corrosion, radiation buildup, and pitting at shutdown. The staff will evaluate the issue of HWC use during its review of an individual application for FDA/DC.

The staff concludes that these requirements are acceptable and their implementation will ensure that the BWR water chemistry is compatible with the selected materials.

PWR Water Chemistry Design Basis

Section 5.5.2.4 of Chapter 1 requires that the water chemistry design basis for PWR plant systems be in accordance with EPRI NP-5960, "PWR Primary Water Chemistry Guidelines," Revision 1, and its subsequent revisions; EPRI NP-6239,"PWR Secondary Water Chemistry Guidelines," Revision 2, and its subsequent revisions; and as supplemented by the guidelines in Table 1.3 in this chapter.

The staff concludes that these requirements are acceptable and their implementation will ensure that the PWR water chemistry is compatible with the selected materials. However, the applicant for FDA/DC or a combined license should reference EPRI NP-7077, Revision 2, instead of EPRI NP-5960, Revision 1. The staff will evaluate this matter during its review of an individual application for FDA/DC or a combined license.

5.6 Conclusion

The staff has reviewed the general materials, materials fabrication, and water chemistry requirements in Section 5 of Chapter 1 of the Evolutionary Requirements Document. These requirements supplement the regulatory guidance in the applicable Standard Review Plan Sections 4.5.1 ("Control Rod Drive Structural Materials"), 4.5.2 ("Reactor Internal and Core Support Materials"), 5.2.3 ("Reactor Coolant Pressure Boundary Materials"), 5.3.1 ("Reactor Vessel Materials"), 5.3.3 ("Reactor Vessel Integrity"), 5.4.2.1 ("Steam Generator Materials"), 6.1.1 ("Engineered Safety Features Materials"), and 10.3.6

("Steam and Feewater System Materials") and, therefore, are acceptable. The implementation of these requirements will ensure that the affected nuclear power plant components will perform in service as designed.

Table 1.2 BWR Water Chemistry Guidelines

Parameter	RWCS	CTSI	MWSE	ICST
Water Quality				
Chloride (parts per billion (ppb))	4.0	4.0	4.0	20.0
Sulfate (ppb)	4.0	4.0	4.0	20.0
Conductivity at 25 °C(μS/cm)	0.075	0.075	0.095	0.3
Silica (ppb as SiO_2)	10.0	-	10.0	20.0
pH at 25 °C				
(minimum)	-	-	6.5	6.2
(maximum)	-	-	7.5	8.0
Corrosion product (ppb)				
Insoluble iron	-	20.0	-	-
Total copper	-	2.0	-	-
All other metals	-	8.0	-	-
Total	3.0	30.0	10.0	20.0
Dissolved oxygen (ppb)				
(minimum)	-	-	-	-
(maximum)	-	20.0	-	-

Notes: Reactor Water Cleanup System (RWCS) effluent during shutdowns and fuel pool/suppression pool cleanup system effluent follow the same guidelines as those for the demineralized water storage tank specified in EPRI NP-4947-SR.

RWCS - reactor water cleanup system during power operation
CTSI - condensate treatment systems influent
MWSE - makeup water systems effluent
ICST - influent to condensate storage tank

Table 1.3 PWR Water Chemistry Guidelines

Water quality parameter	MWST	MWSGs
pH (minimum)	-	7.0
(maximum)	-	7.5
Conductivity at 25 °C (μS/cm)	0.2	0.1
Sodium (parts per billion (ppb))	-	<3.0
Silica (ppb)	-	10
Oxygen (parts per million (ppm)) maximum	0.100	-
Chloride (ppm) maximum	0.15	-
Fluoride (ppm) maximum	0.15	-
Suspended solids (ppm) maximum*	1.0	-
Boric acid (ppm)	-	-
Lithium (ppm)	-	-
Sulfur as sulfate (ppm)	-	-

*Concentration of solids is determined by filtration through a filter with a pore size of 0.45 micron.

Notes:

MWST - makeup water storage tank
MWSGs - makeup water to steam generators

6 RELIABILITY AND AVAILABILITY

Section 6 of Chapter 1 of the Evolutionary Requirements Document gives reliability and availability requirements for safety- and non-safety-related structures, systems, and components (SSCs). Section 6.1 is an introduction to this section. Section 6.2 contains the power production availability requirements and the methodologies, specific strategies, and system features aimed at enhancing reliability for non-safety-related SSCs. Section 6.3 states that quantitative reliability and availability requirements and an analysis process analogous to or integrated with the non-safety-related requirements would be satisfactory for safety-related SSCs. The staff reviewed Section 6 of Chapter 1 of the Evolutionary Requirements Document through Revision 3.

Review Criteria

The NRC identified the need for a safety-oriented reliability effort for the nuclear industry in Section II.C.4 of NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 [Three Mile Island Unit 2] Accident," dated August 1980. Initial NRC research in the area of reliability assurance began in the early 1980s. The results of this research showed that an operational reliability program, based on a feedback process of monitoring performance, identifying problems, taking corrective actions, and verifying effectiveness of the actions, was needed and that other NRC initiatives (e.g., maintenance inspection, performance indicators, aging programs, and technical specification improvement) would address this need. The overall conclusion of this research was that an operational reliability program could be implemented most effectively in performance-based, nonprescriptive regulation, where NRC mandates the level of safety performance to be achieved. For example, licensees could be required to set availability/reliability targets for selected systems and to measure performance compared to the targets. The TMI task was closed out in October 1988 without further action because several NRC initiatives effectively subsumed the operational reliability program effort. The NRC initiatives that formed the basis for closing out this TMI task included efforts to (1) improve maintenance and better manage the effects of aging, (2) improve technical specifications, (3) develop and use plant performance indicators, and (4) develop an operational reliability program as an acceptable means of meeting the station blackout rule (10 CFR 50.63).

In NUREG-1070, "NRC Policy on Future Reactor Designs," dated 1985, the staff recommended the use of a systems reliability program to ensure that the reliability of components and systems important to safety would remain at a sufficient level. To ensure that reliability objectives will be met and to prevent degradation of reliability during operation, it was envisioned that the probabilistic risk assessment performed at the design stage would be used as a tool in making detailed design decisions affecting procurement, testing, and the formulation of operations and maintenance procedures.

In a few specific instances, the NRC is studying or has established reliability targets for systems and components. For example, SRP Section 10.4.9, "Auxiliary Feedwater System," requires that an acceptable auxiliary feedwater (AFW) system design have an unreliability in the range of 1.0E-4 to 1.0E-5 per demand. Generic Issue B-56, "Diesel Reliability," involves efforts to determine, monitor, and maintain emergency diesel generator reliability

levels. Additional regulatory basis for key elements of a reliability assurance program (RAP) can be found in 10 CFR Part 50, Appendix A, and 10 CFR 50.65.

In SECY-89-13, "Design Requirements Related to the Evolutionary Advanced Light Water Reactors," dated January 19, 1989, the staff identified several issues for ALWRs that may go beyond present acceptance criteria defined in the SRP. The RAP, as discussed in SECY-89-13, involved the need for a program to ensure that the design reliability of safety-significant systems, structures, and components is maintained over the life of a plant. In SECY-89-13, the staff informed the Commission that a RAP would be required for ALWR FDA/DC. In November 1989, potential applicants for FDA/DC were informed by letter that "the NRC staff was considering matters that went beyond the current Standard Review Plan...that [the NRC] expects these advanced reactor designs to embody." Reliability assurance was identified as one of these matters.

The RAP can be seen as a program that consists of two distinct parts: the first part, referred to as the "D-RAP (design RAP)", is the responsibility of the designer and applies to vendor submittals for FDA/DC; the second part, referred to as the "O-RAP (operational RAP)", is the responsibility of and applies to an applicant for a combined construction and operating license (combined license). At the design stage, the D-RAP involves a top-level program that defines the scope, conceptual framework, and essential elements of an effective RAP. The D-RAP also implements those aspects of the program that are applicable to the design process. In addition, the D-RAP identifies the relevant aspects of plant operation, maintenance, and performance monitoring for the risk-significant SSCs for the owner/operator's con-sideration in developing the site-specific O-RAP.

The staff's position on the RAP is that a designer's submittal for FDA/DC pursuant to 10 CFR Part 52 would include, in part, the framework for a RAP and would also implement those elements of the RAP that would be applicable during the design phase. In turn, the designer would provide the framework of a RAP to a combined license applicant. This applicant would augment the designer's RAP to reflect plant-specific information and implement those elements applicable during the construction and operations phases. The staff's evaluation was based on guidance contained in the supporting documentation for TMI Task II.C.4, "Reliability Engineering," and SECY-89-13.

Scope of Evaluation

Section 6 of Chapter 1 addresses EPRI-proposed requirements for reliability and availability and provides the supporting rationale for each requirement. In its evaluation, the staff considered the requirements as well as the accompanying rationale of the Evolutionary Requirements Document through Revision 3. In addition, in its letter dated January 9, 1992, EPRI proposed significant modifications to Section 6 of Chapter 1. References to sections of the Evolutionary Requirements Document in this evaluation refer to those in the document as superseded by those proposed by EPRI in the January 9, 1992, letter. In making a safety determination, the staff applied the same acceptance criteria to both the requirements and the rationale. If an EPRI-proposed requirement and its accompanying rationale did not conflict with existing NRC requirements, the staff found that requirement and rationale acceptable.

The Evolutionary Requirements Document primarily addresses D-RAP requirements; only a limited number of requirements in the Evolutionary Requirements Document apply to O-RAP. The O-RAP requirements will need to be provided by an individual applicant for a combined license.

The staff limited its evaluation to only those requirements that could affect plant safety. For example, the staff did not evaluate the merits of economic considerations such as EPRI's plant availability or outage duration goals that are specified in the Evolutionary Requirements Document. It did, however, evaluate the general relationship between safety and economic requirements.

The staff's evaluation of EPRI's requirements resulted in two types of findings: (1) requirements that are acceptable as written and (2) those that are acceptable, but an applicant will be required to provide additional information or guidance. For the requirements of the second type, the staff has provided clarification regarding what additional information a plant designer or an applicant for a combined license should provide when making a submittal that references the Evolutionary Requirements Document.

6.1 D-RAP Goals and Objectives

Section 6.1 of Chapter 1 of the Evolutionary Requirements Document contains the D-RAP goals and objectives. In its letter dated October 10, 1991, the staff stated that a RAP should contain and define the basic framework (scope, purpose, and objective). In its letter dated January 9, 1992, EPRI stated that the scope of a D-RAP was defined in Section 6 of Chapter 1 of the Evolutionary Requirements Document and the D-RAP goals and objectives were defined in Section 6.1. The staff has reviewed EPRI's response and concludes that the definitions of D-RAP scope, purpose, and objective contained in Sections 6 and 6.1 do not conflict with any NRC requirements and are acceptable. The staff also concludes that, in addition to meeting the EPRI-proposed requirements in this section, a plant designer should provide the organizational description and specify organizational accountability for implementing the D-RAP. As an example, Section 3.1.2 of Chapter 10 of the Evolutionary Requirements Document specifies requirements that should ensure that the man-machine interface systems (M-MIS) design is coordinated and implemented properly. An application for FDA/DC should contain an organizational description and accountability for the RAP in similar depth and detail as that in Section 3.1.2 of Chapter 10 of the Evolutionary Requirements Document. The staff will evaluate this description during its review of an individual application for FDA/DC.

6.2 Basic Program Elements of the D-RAP

Section 6.2 of Chapter 1 of the Evolutionary Requirements Document describes the basic program elements required to be included in the D-RAP. In its letter dated October 10, 1991, the staff stated that a RAP should (1) contain and define the program elements and describe how the elements would be applied to the plant structures, systems, and equipment and (2) contain reliability engineering techniques used during the design phase to ensure that the overall design reliability and availability goals are met. In its letter dated January 9, 1992, EPRI proposed a revision to Section 6.2 that describes the requirements for the basic program elements and reliability engineering techniques. The staff's evaluation of this proposed revision is given below.

Analysis Requirements

Section 6.2.1 of Chapter 1 requires an analysis that shows the adequacy of the plant system designs and the recommended maintenance activities, spare parts, surveillance tests, and test intervals needed to support the SSC reliability and availability assumptions of the probabilistic risk assessment (PRA). This analysis will be consistent with the PRA. Further, the reliability and availability analyses will be carried out as an integral part of the design process to influence the design options and allow appropriate cost/benefit tradeoffs during the design of the standard ALWR plant. The Evolutionary Requirements Document requires that this analysis be performed by the plant designer sufficiently ahead of procurement and construction to minimize the impact of potential design changes and ensure that SSC reliability assumptions are met. The staff concludes that these requirements do not conflict with existing NRC requirements and are acceptable. It also concludes that the EPRI requirements in this section should also apply, in addition to PRA methodologies, to deterministic and other methodologies used for making decisions about the adequacy of plant system designs. Furthermore, the staff encourages a vendor or combined license applicant to include references to the analytical methods or models that were used in performing the analyses required by this section in the top-level RAP program document. As an example, Section 3.5.4 of Chapter 10 of the Evolutionary Requirements Document describes analytical methods for use for the M-MIS design. The staff concludes that these or similar methods should be used for all reliability/maintainability analyses performed by the plant designer or combined license applicant. The staff will evaluate this issue during its review of an individual application for FDA/DC or a combined license.

Application of Risk Significance

Section 6.2.2 of Chapter 1 requires the plant designer to ensure that the dominant modes of failure identified by the PRA are appropriately addressed in the plant design consistent with their risk significance. This section further states that the PRA methodology used by the plant designer will provide for importance-weighting of SSCs according to their risk significance and for identifying the dominant failure modes of these SSCs. The staff concludes that these requirements do not conflict with existing NRC requirements and are acceptable. It further concludes that, in addition to PRA methodology, deterministic or other methods may be used in establishing dominant failure modes and risk significance for the RAP. As discussed above, the staff will evaluate this issue during its review of an individual application for FDA/DC or a combined license.

Nuclear Power Plant Reliability Data System Engineering Data Base

Section 6.2.3 of Chapter 1 requires the plant designer to supply the initial input information necessary to establish the nuclear power plant reliability data system (NPRDS) engineering data base for the plant. The staff concludes that the EPRI-proposed requirements do not conflict with existing NRC requirements and are therefore acceptable. However, this section limits the reliability data base to NPRDS. The staff does not want to preclude the use of other data bases that may have merit or that can be shown to be equivalent to current accepted practices. The staff would expect that, in addition to NPRDS data, other reliability data bases could also be considered by plant designers or combined license applicants if they provide the necessary reliability

information. An example may be the data base that results from meeting the requirements of Section 6.2.1.1 of Chapter 1. The staff will evaluate the data bases considered during its review of an individual application for FDA/DC or a combined license.

Reliability Activity Task Descriptions

Section 6.2.4 of Chapter 1 requires the plant designer to provide the owner/operator with descriptions of recommended reliability activities. These activities will include those tasks necessary to maintain SSC environmental qualification, prevent predictable failures, or maintain functional reliability. The plant designer will also recommend appropriate condition monitoring parameters to be periodically evaluated and their acceptable limits to provide added assurance of operability. Condition monitoring recommendations will include provisions for detecting age-related degradation where appropriate. The staff concludes the requirements of this section are acceptable; however, additional guidance that references other areas of the Evolutionary Requirements Document would be beneficial to users of this document. As an example, reliability, maintainability, and testability analyses are described in Chapter 10, Sections 3.5.4, 3.6.1, and 3.6.2, of the Evolutionary Requirements Document. A plant designer should provide similar descriptions to a combined license applicant for all reliability tasks.

In Section 6.2.B.4 of the DSER for Chapter 1, the staff expressed the concern that, as nuclear plants age, the accumulative effect of neutron radiation on the reactor pressure vessel beltline materials will lead to material embrittlement such that materials exhibit loss of fracture toughness and strength. The accumulative high temperature and thermal cycling of the primary coolant systems generate thermal fatigue to materials that causes loss of strength and fracture toughness.

As a result, EPRI added the following statement to Section 6.2.4.5 of Chapter 1 that lists known failure mechanisms: "Loss of strength and/or fracture resistance because of change(s) in metallurgical state of alloys resulting from exposure to high temperature, thermal cycling and/or high radiation." The staff concludes that this modification addresses the staff's concern and, therefore, this DSER confirmatory issue is closed.

Control of PRA Design Assumptions

Section 6.2.5 of Chapter 1 requires the plant designer to provide a program for verifying that PRA assumptions remain within limits required to maintain reliability goals during the design and construction process. This program will apply to all participants in the design and construction process whose activities could affect the plant designer's PRA assumptions. The Evolutionary Requirements Document further states that the results of the verification program will be included as part of the final as-built design documentation. The staff concludes the requirements of this section do not conflict with existing NRC requirements and are acceptable.

6.3 D-RAP Performance Standards

Section 6.3 of Chapter 1 contains the quantitative reliability and availability performance standards for use in the design phase. In its letter dated October 10, 1991, the staff stated that a RAP should contain overall reliability and availability design goals based on safety requirements that have core damage frequencies or probabilities associated with them. In its letter dated January 9, 1992, EPRI proposed to modify Section 6.3 of Chapter 1 to specify D-RAP performance standards. The staff's evaluation of this proposed revision is given below.

Core Damage Frequency

Section 6.3.1 of Chapter 1 states that the plant designer will evaluate the mean annual core damage frequency of the design using PRA and will confirm that this frequency is less than or equal to 10E-5 event per reactor-year, including both internal and external events (excluding sabotage). In Section 2.3.3 of this chapter, the staff discusses EPRI's public safety goal and concludes that the use of this goal is acceptable. However, it will not use EPRI's goal as an acceptance criterion.

The staff concludes that the contributions of the SSCs to the core damage frequency should be apportioned for this, or any other, performance standard. Therefore, the applicant's submittal for FDA/DC should include apportionment of these contributions to the core damage frequency. The staff will evaluate the adequacy of the apportionment of the SSCs' contribution to core damage frequency during its review of an individual application for FDA/DC.

Inadvertent Depressurization

Section 6.3.2 of Chapter 1 states that non-safety-related active reactor coolant system (RCS) makeup capability and any other necessary measures will be provided so that the chance of inadvertent RCS depressurization can be demonstrated by reliability analysis to be less than 10 percent over the entire 60-year life of the plant. Further, recovery from inadvertent RCS depressurization will be rapid enough that the lifetime average design availability requirements can still be met assuming one inadvertent RCS depressurization during the 60-year plant life. The staff notes that referring to the active RCS makeup capability as non-safety related is inconsistent with risk-significant concepts in Section 6.2.2 of Chapter 1 of the Evolutionary Requirements Document. The staff concludes that these requirements do not conflict with existing NRC requirements and are, therefore, acceptable. However, in addition to these requirements, an FDA/DC applicant should explicitly state the priority of safety in accident recovery. The staff will evaluate this matter during its review of an individual application for FDA/DC.

Production Availability Requirement

Section 6.3.5 of Chapter 1 states the plant will be designed for an annual average production availability of more than 87 percent over its life. The staff's review of this requirement was limited to areas where economic considerations could potentially compromise plant safety. The staff concludes that this requirement does not conflict with existing NRC requirements and is therefore, acceptable. However, an FDA/DC or combined license applicant

should explicitly discuss the relationship between safety and production availability in a D-RAP or O-RAP when addressing this design requirement. The staff will evaluate this matter during its review of an individual application for FDA/DC or a combined license.

Refueling Duration Capability

Section 6.3.7 of Chapter 1 states that the plant will be designed so that the total duration of a no-problem refueling outage will be 17 days or less (breaker to breaker) assuming 24-hour productive days. The staff concludes that this requirement does not conflict with existing NRC requirements and is acceptable. However, a D-RAP or O-RAP submittal from an FDA/DC or combined license applicant should explicitly state that plant safety will not be compromised in attempting to satisfy this EPRI design requirement. The staff will evaluate this matter during its review of an individual application for FDA/DC or a combined license.

Planned Outages

Section 6.3.8 of Chapter 1 states that the plant will be designed so that refueling and regular maintenance will be completed in an average of less than 25 days per year. An average of 25 days per year for refueling and plant maintenance corresponds to 50 days in a 24-month fuel cycle. The staff concludes that this requirement does not conflict with existing NRC requirements and is acceptable. However, a D-RAP or O-RAP submittal from an FDA/DC or combined license applicant should explicitly state that plant safety will not be compromised in attempting to satisfy this EPRI design requirement. The staff will evaluate this matter during its review of an individual application for FDA/DC or a combined license.

Major Outages

Section 6.3.9 of Chapter 1 states that the plant will be designed so that the frequency and duration of major outages will not exceed 180 days per 10 years. The staff concludes that this requirement does not conflict with existing NRC requirements and is, therefore, acceptable. However, a D-RAP or O-RAP submittal from an FDA/DC or combined license applicant should explicitly state that plant safety will not be compromised in attempting to satisfy this EPRI design requirement. The staff will evaluate this matter during its review of an individual application for FDA/DC or a combined license.

6.4 System Design

Section 6.4 of Chapter 1 of the Evolutionary Requirements Document describes the qualitative processes to be used in system design. In its letter dated October 10, 1991, the staff stated that a RAP should establish a set of reliability and availability goals at the system level to ensure that the reliability and availability goals for the overall design are met. In its letter dated January 9, 1992, EPRI proposed to modify Section 6.4 of Chapter 1 to specify general system design goals for a RAP.

Shared Systems

Section 6.4.1 of Chapter 1 requires that, for multiple-unit plants on a single site, the number of shared systems will be limited to auxiliary support systems such as sewer, auxiliary steam, or site security. For any systems that are shared, the plant designer or combined license applicant will analyze the effect of any failure or any testing in that system that will affect the maintenance, ALARA (as low as is reasonably achievable) goals, availability, safety, or operability of other systems and the availability of each unit. The staff concludes that these requirements do not conflict with existing NRC requirements and are, therefore, acceptable.

Startup and Testing

Section 6.4.2 of Chapter 1 requires that the plant designer or combined license applicant review and optimize the startup testing program for initial startup and for startup following refueling/maintenance outages to support completion of required testing with a minimum impact on the availability of the plant. As a basis for this review and optimization, the plant designer will review existing LWR startup test programs and any available reports on optimization of these test programs. The staff concludes that these requirements do not conflict with existing NRC requirements and are, therefore, acceptable.

Failure Mechanisms

Section 6.4.3 of Chapter 1 states that the plant designer will ensure that the systems, equipment, and plant design will account for failure mechanisms shown to have a significant effect on downtime. This section further states the ALWR will be designed so that known failure mechanisms will not prevent the ALWR from achieving its design life or meeting the availability and event frequencies as described in Section 6.3 of Chapter 1. The staff concludes that these requirements do not conflict with existing NRC requirements and are, therefore, acceptable.

Specific System Design Features

Section 6.4.4 of Chapter 1 states that the plant designer will incorporate design features to support high reliability of safety-significant SSCs and high plant availability. The staff concludes that this requirement does not conflict with existing NRC requirements and is, therefore, acceptable.

Minimum Number of Components

Section 6.4.5 of Chapter 1 states that the plant designer will reduce the number of active components required to meet the intended function of operability and maintainability. The staff concludes that this requirement does not conflict with existing NRC requirements and is, therefore, acceptable.

6.5 Conclusion

The staff's overall conclusion is that the requirements in Section 6 of Chapter 1 of the Evolutionary Requirements Document are directed at the plant designer and the D-RAP and very few requirements are directed toward the

procurement, construction, and operational phases, which are the responsibility of the combined license applicant. Additional guidance and requirements will be necessary to support the development of a combined license applicant's O-RAP.

The staff concludes that the reliability and availability requirements in the Evolutionary Requirements Document are consistent with accepted industry practices and principles and do not conflict with existing regulatory requirements and guidelines. If the information identified in the Evolutionary Requirements Document and the additional information identified in this report are provided by a plant designer, the D-RAP submittal should be acceptable.

7 CONSTRUCTION AND CONSTRUCTIBILITY

Section 7 of Chapter 1 of the Evolutionary Requirements Document specifies minimum requirements and goals proposed by EPRI with regard to the construction of an ALWR. EPRI states that these requirements support the overall objectives described in Volume I of the Requirements Document, particularly by reducing the utility's risk and plant capital costs through an improved construction schedule, greater productivity, and better construction techniques. This section discusses construction schedule goals; integration of design, construction, and startup schedules; use of computer technology for design, construction, and startup activities; and integration of system completion, testing, and owner/operator acceptance.

Many of EPRI's requirements concerning construction and constructibility of the plant are beyond the regulatory purview of the NRC. The staff evaluated these requirements only from the aspect of how they may adversely affect the safe design, construction, and operation of the plant. However, it did identify items that are part of its regulatory responsibility.

Inspection of Construction Activities

The NRC has the statutory responsibility, regardless of construction schedule, to verify that the plant is constructed in accordance with the design documents tendered with the application for an operating license. The owner/builder must ensure that construction activities permit verification of the acceptability of the plant configuration in accordance with the requisite NRC Inspection Manual chapters. The staff will evaluate this matter during its review of an individual application for a combined license.

In Section 7.2.C.2 of the DSER for Chapter 1, the staff indicated that EPRI had committed to modify its data base program to include milestones for scheduling construction verification inspection points for inspection and enforcement personnel and startup tests. EPRI included this information in Section 7.2.8.1.3 of Chapter 1 of Revision 1 of the Evolutionary Requirements Document. Therefore, this DSER confirmatory issue is closed.

Schedular and Administrative Requirements

The schedular and administrative requirements proposed by EPRI in this section are outside the staff's regulatory purview. The staff notes that these measures, although outside the scope of the NRC's review, should assist the owner/builder's configuration control to facilitate preservation of the licensing basis.

Quality Assurance

In the DSER for Section 7 of Chapter 1, the staff identified an open issue that construction activities for which the owner/builder must provide a site organization should include quality assurance and quality control. This issue resulted from the staff's review of Revision 0 of the Evolutionary Requirements Document.

In response to the staff's concern, EPRI revised Section 7.2.8.2 of Chapter 1 to make it clear that the site organization plan must include provisions for quality control and quality assurance. The staff concludes that this revision resolves the staff's original concern and, therefore, this DSER open issue is closed.

However, EPRI added additional information that was not clear in regard to its intent regarding quality assurance. For example, Section 7.2.8.2 of Chapter 1 could be interpreted as not requiring a commitment to apply pertinent quality assurance program requirements to facilities and systems that have some importance to safety even though they are not safety related (for example, see the new maintenance rule, 10 CFR 50.65(b)(2)). The staff concludes that pertinent quality assurance provisions should be applied to these activities and items. Applicants for FDA/DC or a combined license will have to acceptably describe a quality assurance program for these activities and items. The staff will evaluate this matter during its review of an individual application for FDA/DC or a combined license.

Physical Security

In Section 7.2.8.2 of Chapter 1, EPRI requires a plant security plan covering the construction phase. Although this is a utility requirement outside the scope of 10 CFR Part 73, the requirement is compatible with NRC requirements.

In Section 7.9.6 of Chapter 1, EPRI requires the utility to establish security boundaries as part of the startup testing program. In a letter dated May 13, 1991, EPRI agreed that the detailed construction and startup schedule will have to address NRC review and approval of the installed security system for the operating phase before the first fuel loading, but that this milestone was beyond the scope of the Evolutionary Requirements Document. The staff, therefore, will address this matter during its review of an individual application for a combined license.

The staff expects that at least 60 days before loading fuel, a combined license licensee will have confirmed that security systems and programs described in its physical security plan, safeguards contingency plan, and guard qualification and training plan have achieved operational status and are available for NRC inspection. Operational status means that the security systems and programs are functioning in their entirety as they would when the reactor is operating and will continue to do so. The licensee's determination that operational status has been achieved must be based on tests conducted under realistic operating conditions of sufficient duration that demonstrate that the equipment is properly operating and capable of long-term, reliable operation; that procedures have been developed, approved, and implemented; and that personnel responsible for security operations and maintenance have been appropriately trained and have demonstrated their capability to perform their assigned duties and responsibilities.

Reliability of Modular Construction

In Section 7.7 of Chapter 1, EPRI proposes the use of modular construction techniques for ALWRs. These techniques would allow construction and testing of portions of the plant in onsite fabrication shops or off site. These

modules would then be assembled in the laydown area of the plant. This technology offers the potential for much shorter construction schedules and reduced costs.

Specific licensing criteria addressing modular construction have not been developed for nuclear power plant construction. Structures, systems, and components that are assembled using modular construction techniques must possess, as a minimum, the same degree of structural strength and reliability as such items provided in currently licensed plants that were constructed using current onsite construction techniques. Items to be considered include segmented rebar cage connections and in-containment steel/concrete sandwich-type shear walls for which there are no modular construction design criteria and for which test information is limited. Other areas of concern include the integrity of joints (including strength and ductility), seismic damping values and stiffness degradation in structural modules, quality assurance and quality control requirements for transportation and installation of modules, and the scope of the verification testing after the modules are installed. The staff will address this matter during its review of an individual application for a combined license, should the applicant propose use of these techniques.

8 OPERABILITY AND MAINTAINABILITY

8.1 Introduction

In Section 8 of Chapter 1 of the Evolutionary Requirements Document, EPRI proposes requirements that are meant to enhance the operability and maintainability of the ALWR by incorporating experience gained at operating facilities. These requirements are intended to minimize the need for maintenance and surveillance, thereby minimizing the dose of radioactivity to plant personnel. Included are general design criteria for the control room and other control locations and criteria for designing the instrument and control systems and factoring in man-machine interface considerations.

8.2 Provisions To Enhance Operability and Maintainability

In Section 8.2 of Chapter 1 of the Evolutionary Requirements Document, EPRI proposes requirements regarding resolution of known operational and maintenance problems, standardization of operating and maintenance procedures and related training, and standardization of components and equipment. This section also gives human factors requirements that are related to the operation of the plant, including consideration of instrumentation and controls, control room design, and environmental conditions. In addition, EPRI provides requirements to address human factors that are related to maintenance of the plant, including support systems, accessways, and orientation of equipment. EPRI specifies requirements to integrate operations and maintenance requirements and addresses preventive maintenance and inspection concerns. Section 8.2 specifies requirements for the qualifications, organizations, and training of operations and maintenance personnel.

Preventive Maintenance and Inspections

In Section 8.2.C.2 of the DSER for Chapter 1, the staff recommended that specific codes and standards relating to inservice inspection and test requirements be included in the Evolutionary Requirements Document. In response, in Section 8.2.6.3 of Chapter 1, EPRI requires that the preventive maintenance programs include the integration of equipment inspection and testing requirements imposed by the codes and standards that apply at the time of design. Also, EPRI specifies 10 CFR 50.55a in Table B.1-1 of Appendix B to Chapter 1, which requires using Section XI of the ASME Code and addenda for the inservice inspection requirements. The staff concludes that these requirements for the preventive maintenance program follow the ASME Code and are acceptable. Therefore, this DSER confirmatory issue is closed.

Radiation Exposure

Section 8.3.2 of Chapter 1, to reduce worker radiation exposure, EPRI states that temperature and humidity in radiation areas will be controlled by the heating and ventilation system. All plant areas will be adequately illuminated to minimize the time and exposure associated with the installation of temporary lighting in work areas. To reduce worker doses, equipment will be oriented to facilitate maintenance operations. These provisions are intended to maintain worker doses as low as is reasonably achievable (ALARA) and, therefore, are acceptable.

Personnel and Staffing

In the DSER for Section 8 of Chapter 1, the staff stated that EPRI had indicated that it would add a statement to the Evolutionary Requirements Document regarding the qualification of plant operating personnel. In its letter dated May 5, 1991, EPRI stated that qualifications and training requirements for the plant operating personnel were the responsibility of the plant owner and were outside the scope of the Evolutionary Requirements Document. EPRI added Section 8.2.6.5, which indicates that these personnel will meet the requirements of ANSI/ANS 3.1, "Selection and Training for Nuclear Power Plants." ANSI/ANS 3.1 excludes the selection and training of site security personnel, the staff concludes that there are no conflicts with the security training requirements of 10 CFR 73.55(b). The staff concludes that this is in accordance with Regulatory Guide 1.28, "Quality Assurance Program Requirements," and is acceptable. Therefore, this DSER confirmatory issue is closed.

Acoustical Monitoring

In Section 8.2.B of the DSER for Chapter 1, the staff recommended that appropriate noise level requirements concerning acoustical monitoring requirements be added to Chapter 1 because noise is a major contributor to operator fatigue and ineffectiveness. In Revision 1 of Section 8.2.4.4.3 of Chapter 1, EPRI added noise level requirements that ensure that the designer of the plant will consider both reduction and attenuation of noise sources to reduce noise exposure of operators to levels specified by the Occupational Safety and Health Administration (OSHA). The staff did not find the revision acceptable because OSHA standards reflect maximum permissible exposures and not necessarily the sound levels required for operations and maintenance personnel to perform their jobs. By letter dated March 19, 1992, EPRI revised the statement to read:

> The design of the plant shall consider both reduction and attenuation of noise sources to reduce exposure to operation and maintenance personnel. Personnel shall be provided with acoustical environments which will not cause personal injury, interfere with voice or other communications, cause fatigue, or degrade overall system effectiveness.

The staff concludes that the statement is acceptable. Therefore, this DSER confirmatory issue is closed.

Human Factors Considerations

In Section 8.2.B.4 of the DSER for Chapter 1, the staff recommended that IEEE P1023-1988, "Guide for the Application of Human Factors Engineering to Systems, Equipment, and Facilities of Nuclear Power Generating Stations," and EPRI-2360, "Human Factors Methods for Assessing and Enhancing Power Plant Maintainability," be referenced in this section. Although EPRI has not referenced these documents in the Evolutionary Requirements Document, the staff will evaluate an individual application for FDA/DC or a combined license to ensure that these documents have been considered. This DSER open issue is closed.

60-Year Life

Section 8.2.3.3 of Chapter 1 states that components will be designed for an operating period of 60 years (minimum) or in accordance with Section 11.3 of Chapter 1 of the Evolutionary Requirements Document. As discussed in Section 3.3 of this chapter, the staff will review the ALWR designs for a 60-year life notwithstanding the fact that a 40-year limit is specified in the Atomic Energy Act and NRC's regulations.

8.3 Minimizing Dose Levels to Personnel

Section 8.3 of Chapter 1 of the Evolutionary Requirements Document specifies the general requirements and design features that will be used to minimize radiation levels and to maintain personnel doses ALARA. Specific design criteria are addressed in other chapters of the Evolutionary Requirements Document, primarily Chapter 6.

To achieve its dose goal of 100 person-rem per year for the ALWR, EPRI established several basic requirements that must be met. These include the control of materials selection (including the elimination of cobalt to the extent possible); the use of adequate temporary and permanent shielding; the design of components to permit cleaning and chemical decontamination; the application of robotics for cleanup, maintenance, and inspection tasks; the packaging of equipment in modules for rapid disassembly for inspection and maintenance; and the design of heating and ventilation systems to control temperature and humidity in radiation work areas.

In Section 8.3.3 of Chapter 1, EPRI states that the plant designer will consider the use of electropolished surfaces for those areas of the plant (e.g., large-diameter reactor coolant system piping, steam generator channel heads) where this treatment will significantly reduce the dose to personnel during maintenance.

Section 8.3.4 of Chapter 1 states that the plant designer will perform an analysis to determine the effectiveness of using robotic applications in the ALWR. Inspection and surveillance functions will include reading of instruments and gauges, performing radiation surveys and measuring radiation levels, and taking smear surveys. Maintenance functions will include steam generator inspection and maintenance, control rod drive removal, radwaste drum handling, spent fuel consolidation, equipment decontamination, and routine surveillance and maintenance tasks. The Evolutionary Requirements Document specifies that the ALWR will include design features such as wider doors and aisles, ramps, and modular construction of equipment and systems (for ease of equipment removal and replacement) to facilitate the use of robotic devices. Table 1.8-4 includes "verify security locks" as one of several functions to be evaluated by the plant designer as a candidate for robotic inspection and surveillance. However, in its letter of May 17, 1991, EPRI stated that details of the security functions to be performed and replacement of a security officer were outside the scope of the Evolutionary Requirements Document. The staff, therefore, will address this matter during its review of an individual application for a combined license.

The staff concludes that the design features in this section are intended to minimize dose levels to plant personnel, are in compliance with Regulatory Guide 8.8, and are, therefore, acceptable.

8.4 Facility Requirements

Section 8.4 of Chapter 1 of the Evolutionary Requirements Document specifies that the plant will be designed to provide adequate support facilities for personnel and equipment. This section addresses general requirements for controlling personnel access to the plant and, particularly, to radioactive work areas. It provides requirements for plant services, contaminated and clean work shops, and spare parts control and includes requirements for the design of the plant's personnel access portal.

In its letter dated May 13, 1991, EPRI committed to change some of the security terms in Sections 8.4.1.1 and 8.4.1.2 of Chapter 1 to be consistent with the terminology in Section 5 of Chapter 9 of the Evolutionary Requirements Document. The staff has verified that these changes have been acceptably included in EPRI's June 11, 1991, markup.

EPRI states that the ALWR will be designed to allow controlled access to the plant for the number of personnel needed to perform the required activities. Changing rooms will be located away from radiation sources and will have lockers for at least 1,000 people. Radioactive work areas will be separated from clean work areas. To the extent possible, the ALWR design will incorporate ramps or steps instead of ladders for personnel movement between floors in areas where personnel are required to wear anticontamination clothing. Such measures are intended to facilitate the processing and movement of large numbers of personnel through the plant during major maintenance outages. The staff concludes that these features do not conflict with the Commission's regulations and guidance and are, therefore, acceptable.

8.5 Provisions for Replacement of Major Components

Section 8.5 of Chapter 1 of the Evolutionary Requirements Document specifies requirements to facilitate the replacement of all major components other than the reactor vessel and basic plant structures. The plant design will also include plans for the transportation and storage of major plant components that may be contaminated and contaminated special tools and equipment removed from the buildings. Designing the plant so that major components can be easily removed and transported out of the buildings without major structural modifications will shorten plant outage time and will, therefore, result in lower overall personnel exposures. The staff concludes that these features do not conflict with the Commission's regulations and guidance, and are, therefore, acceptable.

8.6 Inspection and Testing

In Section 8.2.B of the DSER for Chapter 1, the staff requested that a qualifying statement be added to Chapter 1 specifying that the plant designer will select pumps and valves that can meet the requirements of Sections IWP and IWV of ASME Code, Section XI, which contain rules for inservice testing of pumps and valves. In response to this request, Revision 1 of Section 8.6.1 of Chapter 1 was added to provide this commitment. Therefore, this DSER confirmatory issue is closed. However, a more detailed discussion by the staff of inservice testing is contained in Section 12 of this chapter.

Operation Problem Areas

In the section entitled "LWR Operation Problem Areas To Be Addressed in ALWR Design" of the DSER for Chapter 1, the staff requested that EPRI add the following item to Table 8-2 of the Evolutionary Requirements Document: "Insufficient structural integrity and mechanical reliability of pump components." This item was added to the renumbered Table 1.8-2 in Chapter 1. Therefore, this DSER confirmatory issue is closed.

8.7 Hazardous and Toxic Chemicals

Section 8.7 of Chapter 1 of the Evolutionary Requirements Document specifies that the use of hazardous and toxic chemicals in the plant will be minimized to the extent practicable. This section excludes the use of hazardous chemicals in radiation-controlled areas, unless there is no practical alternative to their use. Requirements for the use of such substances are also provided. The staff concludes that these requirements do not conflict with the Commission's regulations and guidance and are, therefore, acceptable.

8.8 Conclusion

The staff concludes that the requirements of Section 8 of Chapter 1 of the Evolutionary Requirements Document do not conflict with the Commission's regulations and are, therefore, acceptable. However, certain details regarding human factors engineering and physical security were insufficient to enable a final determination regarding compliance with regulatory guidance. Therefore, the staff will review an individual application for FDA/DC to ensure that this guidance has been met.

9 QUALITY ASSURANCE

Section 9 of Chapter 1 of the Evolutionary Requirements Document identifies the major elements of an overall quality assurance (QA) program for an evolutionary ALWR plant. EPRI specifies the requirements for the supporting QA programs for the primary organizational entities involved in and supporting design, procurement, construction, and preoperational testing of an evolutionary ALWR plant. EPRI has also affirmed that this section is intended to be in accordance with all current regulatory requirements for QA at the level of detail provided.

In the DSER for Section 9 of Chapter 1, the staff stated that EPRI's responses to earlier staff comments had not been incorporated into the Evolutionary Requirements Document. EPRI has incorporated the changes in Sections 9.2.3 and 9.2.4 of Revision 1 of Chapter 1. The staff concludes that these sections are in accordance with Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)," and are, therefore, acceptable. Therefore, this DSER confirmatory issue is closed.

In the DSER for Section 9 of Chapter 1, the staff stated that two line items needed to be added to Table 9-1. EPRI has incorporated the requested changes in Table 1.9-1 of Revision 2 of Chapter 1. The staff concludes that this table provides an acceptable list of typical quality and quality assurance problems experienced during the design and construction of currently operating nuclear plants. Therefore, this DSER confirmatory issue is closed.

In the DSER for Section 10 of Chapter 1, the staff stated that the cross-references in Table B.1-1 were incorrect. As discussed by the staff in Section 10 of this chapter, this table has been corrected. In the area of quality assurance, Table B.1-1 lists 10 CFR 50.34(f)(3)(ii), 50.34(f)(3)(iii), 50.55(e), 50.55(f), and 50.55a; General Design Criterion (GDC) 1 of Appendix A to 10 CFR Part 50; and Appendix B to 10 CFR Part 50. In each case, Table B.1-1 indicates that the ALWR will be designed to comply with these requirements and that the lead chapter for this subject is Chapter 1 of the Evolutionary Requirements Document. Table B.1-2 also lists Regulatory Guides 1.26 (Rev. 3), 1.28 (Rev. 3), 1.29 (Rev. 3), 1.30 (Rev. 0), 1.37 (Rev. 0), 1.38 (Rev. 2), 1.54 (Rev. 0), 1.94 (Rev. 1), and 1.116 (Rev. 0-R) and SRP Section 17.1 (Rev. 2). Revision 4 of Table B.1-2 indicates that the ALWR will be designed to comply with this guidance and that the lead chapter for this subject is Chapter 1 of the Evolutionary Requirements Document. The staff concludes that these references are acceptable. Therefore, this DSER confirmatory issue is closed.

In Section 6.1.2 of the DSER for Chapter 10, the staff raised a concern regarding the QA program for software. In its letter dated January 28, 1992, EPRI provided its response to this concern. However, EPRI limits its software QA program to safety-related software. Similarly, as discussed in Section 7 of Chapter 1 of the Evolutionary Requirements Document, EPRI appears to limit its QA to safety-related items. The staff is concerned that the Evolutionary Requirements Document could be interpreted as not requiring a commitment to apply pertinent QA program requirements to software, facilities, structures, systems, and components that have some safety importance or have some importance to safety even though they are not safety related. The staff concludes that pertinent QA provisions should be applied to these activities and items

in accordance with the requirements of GDC 1 of Appendix A to 10 CFR Part 50. Applicants for FDA/DC or a combined license will have to acceptably describe a QA program for these activities and items. The staff will evaluate this matter during its review of an individual application for FDA/DC or a combined license.

The staff concludes that, with the exceptions noted above, the requirements of Section 9 of Chapter 1 of the Evolutionary Requirements Document do not conflict with the Commission's regulations and are, therefore, acceptable.

10 LICENSING

Section 10.1 of Chapter 1 of the Evolutionary Requirements Document specifies EPRI-proposed licensing requirements for future ALWRs. EPRI states that Section 1 of Appendix B to Chapter 1 contains a list of NRC regulations and guidelines currently applicable to LWR design and identifies its position with respect to each. EPRI states that Section 1 of Appendix B to Chapter 1 will represent the regulatory requirements it believes are applicable to the ALWR design at the level of detail consistent with that of the Evolutionary Requirements Document when that document is completed.

Section 10.2 of Chapter 1 states that the ALWR will be designed to comply with the NRC regulatory requirements and guidance in effect on January 1, 1990, consistent with the commitments in Section 1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document. EPRI states that these requirements and guidance include applicable Commission regulations specified in 10 CFR, general design criteria, NRC policy statements, regulatory guides, the Standard Review Plan (NUREG-0800), and other documentation that resolves unresolved and generic safety issues. Although the staff understands EPRI's need to "freeze" the requirements it addresses to those in effect on January 1, 1990, the staff expects that the design certification applications will be in compliance with the Commission's regulations and guidance that are applicable and in effect at the time the certification is issued. The staff will evaluate this compliance during its review of an individual application for FDA/DC.

In addition, issue resolutions that are different from those arrived at during the staff's review of the Evolutionary Requirements Document may be developed as the staff completes its reviews of the detailed design information provided in the applications for FDA/DC and as these designs are subjected to the design certification rulemaking process. Therefore, the staff expects that the ALWR plant designers will comply with the issue resolutions adopted by the NRC staff during its reviews of applications for FDA/DC in accordance with the requirements of 10 CFR Part 52. The staff will evaluate this compliance during its review of an individual application for FDA/DC.

In the DSER for Section 10 of Chapter 1, the staff stated that the cross-references in Table B.1-1 were incorrect. Because EPRI had committed to update the Table, the staff identified this as a confirmatory issue. The staff has verified that EPRI has corrected this table. Therefore, this DSER confirmatory issue is closed. Should EPRI make additional revisions to the Evolutionary Requirements Document, the staff will ensure that Table B.1-1 is upated appropriately.

Section 10.2.4 of Chapter 1 states that the plant designer will provide an ALWR design that is consistent with the disposition of the regulatory requirements and guidance identified in Section 1 of Appendix B. EPRI defines the commitment specified in Appendix B as follows:

- <u>Comply</u>

  The "comply" designation indicates that the ALWR design will comply fully with all regulatory requirements and guidance provided by the reference.

- Optimization Subject

  "Optimization subjects" are EPRI-initiated proposals to deviate from regulatory requirements. EPRI proposes to resolve these issues by providing technically supportable alternatives to current regulatory requirements. EPRI specifies that the ALWR design will comply with all regulatory requirements and guidance for a regulatory item associated with an optimization subject, except as described in Section 2 of Appendix B to Chapter 1 of the Evolutionary Requirements Document.

Section 10.2 of Chapter 1 states that the plant designer will provide a design that is consistent with Sections 2 and 3 of Appendix B to Chapter 1. These sections include EPRI-proposed resolutions for optimization subjects and generic safety issues, respectively.

In the DSER for Section 10 of Chapter 1, the staff indicated that NRC approval of the Requirements Document implies general agreement with the design criteria in the document, but such approval is not meant to imply that the document is a complete and adequate set of requirements for a nuclear power plant. The meaning of NRC's approval of the Requirements Document was listed as an open issue. As EPRI has stated, the Requirements Document is intended for use with companion documents, such as utility procurement specifications, that cover the remaining technical requirements for a specific plant. As discussed by the staff in Section 1 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 design certification for a specific design application, nor is it to be used to substitute for any portion of the staff's review of an individual application for FDA/DC. On the basis of the above discussion, this DSER open issue is closed.

Inspections, Tests, Analyses, and Acceptance Criteria

As stated in 10 CFR 52.47(a)(1)(vi), applications for design certification must include proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that references the design is built and will operate in accordance with the design certification. Section 3.3.5 of Volume I, "Policy and Top-Tier Design Requirements," of the Requirements Document requires the plant designer to prepare a set of ITAAC, with technical basis provided, that will demonstrate that the plant has been constructed and will be operated in conformity with Commission regulations, the combined license, and the Atomic Energy Act. The section also requires the nature and level of detail of acceptance criteria to be such as to allow the NRC staff to verify that the acceptance criteria have been met. Although Section 7.9 of Chapter 1 specifies certain requirements related to system completion and startup testing, the overall requirements in Chapter 1 do not provide guidance regarding the scope and content of ITAAC. In its letter dated May 13, 1991, EPRI indicated that it did not plan to provide ITAAC guidance, but was supporting the development by the Nuclear Management and Resources Council of industry guidance regarding ITAAC for design certification. Each plant designer will submit ITAAC for the specific design for NRC review using the industry's guidance.

The design detail to be developed by an applicant for design certification can be embodied in three parts: (1) Tier 1 information that will be submitted in the application and certified by rulemaking, (2) Tier 2 information that will be submitted in an application but not certified, and (3) information not submitted but available for NRC audit. The two-tiered approach is to standardize design details to the maximum extent practicable, but allow the flexibility needed to finalize the design and construct the facility considering the procurement and design reconciliation process. In SECY-90-377, "Requirements for Design Certification Under 10 CFR Part 52," dated November 8, 1990, the staff states that Tier 1 will include information developed during the conceptual phase, such as design criteria and bases, and certain information developed during the preliminary and detailed design phases, such as descriptions of systems and key components, functional and performance requirements for plant systems, simplified electrical single-line diagrams, simplified piping and instrumentation drawings, general arrangement drawings, and ITAAC.

The staff is developing additional guidelines for the scope and content of ITAAC and is evaluating pilot ITAAC submittals based on the General Electric advanced boiling water reactor design. As described in SECY-91-178, "Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Design Certifications and Combined Licenses," dated June 12, 1991, Tier 1 ITAAC will be at a level of detail corresponding to the Tier 1 design information of the certified design rule. The staff expects that the Tier 1 verification requirements will be high level in nature and will address the design at a system functional performance level of detail. Numerical acceptance criteria values will only be specified when failure to meet the stated acceptance criteria would clearly indicate a failure to properly implement the design. Although including appropriate guidance on scope and content for ITAAC submittals in the Evolutionary Requirements Document would ensure that each design submittal will include a complete and adequate ITAAC package for staff review, ITAAC is clearly the responsibility of the plant designer. The staff will evaluate the proposed ITAAC during its review of an individual application for FDA/DC or a combined license.

11 DESIGN PROCESS

11.1 Introduction

In Section 11 of Chapter 1 of the Evolutionary Requirements Document, EPRI proposes requirements for the design process for an ALWR. This process will include such activities as development, testing, analyses, preparation of specifications and drawings, modeling, documentation, and support of others as required to complete the licensing, construction, and startup of the plant, including turning the plant over to the operator. EPRI states that "licensing" is intended to include compliance with all Federal, State, and local regulations, including environmental impact reports.

11.2 Technology Base

In Section 11.2 of Chapter 1 of this Evolutionary Requirements Document, EPRI states that ALWRs should be designed using systems, components, and equipment proven through several years of acceptable service in LWR plants. EPRI defines proven systems, components, and equipment to be those that have the same characteristics and that use materials proven under the same environmental and working conditions as those that have been successfully applied for at least several years in existing LWRs or similar operating environments and applications. Although the Evolutionary Requirements Document does not preclude the use of advanced technologies, EPRI states that unproven systems or equipment may be used only if sufficient justification is provided to support their use.

The staff concludes that these requirements do not conflict with the Commission's regulations and guidance. In the regulatory departure analysis in Appendix B to this chapter, the staff discusses the process it will use for determining the need for a prototype or other demonstration facility for the ALWR.

11.3 Design Life

Section 11.3 of Chapter 1 of the Evolutionary Requirements Document provides requirements to be used to design for a plant life of 60 years. The staff concludes that these requirements do not conflict with the Commission's regulations and guidance. As discussed in Section 3.3 of this chapter, if requested by the applicant, the staff will review the ALWR designs for a 60-year life, notwithstanding the fact that a 40-year license term limitation is specified in the Atomic Energy Act and NRC's regulations.

In Section 8.2.C.3 of the DSER for Chapter 1, the staff recommended that EPRI require using the experience gained from two pilot plant studies (for Northern States Power and Virginia Electric Power plants) in the plant life extension program sponsored by EPRI and the Department of Energy. Section 11.3.4.5 of Chapter 1 of the Evolutionary Requirements Document appropriately references these studies. Therefore, this DSER confirmatory issue is closed.

11.4 Plant Simplification

Section 11.4 of Chapter 1 of the Evolutionary Requirements Document states that the ALWR design will be simpler than that of current operating plants because a minimum number of mechanical components (valves, pumps, heat exchangers, snubbers) and a minimum of instrumentation and controls will be used. EPRI specifies that the plant will be designed to simplify operations during all modes of operation, including operator actions to diagnose and manage abnormal and accident conditions.

The staff agrees with the overall objective to simplify systems and operations and will ensure that this objective has been implemented in accordance with the Commission's regulations and guidance during its review of an individual application for FDA/DC.

11.5 Standardization

Section 11.5 of Chapter 1 of the Evolutionary Requirements Document states that the ALWR design will be developed as a standard plant design, including, as a minimum, a standard design basis, standard site envelope, standard equipment, and standard technical documentation.

The staff supports the concept of standardization, as can be seen in the promulgation of 10 CFR Part 52, the Commission's standardization policy, and the staff's review of the EPRI Requirements Document. It, therefore, will ensure that this concept has been implemented in accordance with the Commission's regulations and guidance during its review of an individual application for FDA/DC.

11.6 Specific ALWR Design Process Requirements

Section 11.6 of Chapter 1 of the Evolutionary Requirements Document specifies that the plant designer will use design methods required by applicable standards, codes, and regulations, as well as those based on the designer's own experience, methods, and design tools. EPRI specifies that the designer should use the special studies, evaluations, and design approaches that are provided throughout the Evolutionary Requirements Document.

In the regulatory departure analysis in Appendix B to this chapter, the staff discusses the applicable codes and standards to be used in the ALWR designs.

11.7 Project Information Network

Section 11.7 of Chapter 1 of the Evolutionary Requirements Document requires that a project information network (PIN) based on the guidelines and methodology of EPRI NP-5159, "Guidelines for Specifying Integrated Computer-Aided Engineering (CAE) Applications for Electric Power Plants," be established to define the activities, entities, attributes, and relationships for the total plant cycle. EPRI specifies that this network will be used by all ALWR plant design participants in organizing and identifying the products of the design process.

EPRI specifies that the PIN will segment the design into a number of systems and system groups, modeling, maintenance of the technical data base, systematic retrieval of information, standard identification of data, and

provision of a single source of technical data. The staff's evaluation of the PIN and its relationship to the information management system that EPRI specifies is provided in Section 11.12 of this chapter.

11.8 Design Development Plan

Section 11.8 of Chapter 1 of the Evolutionary Requirements Document specifies that the plant designer will develop a plan for the development and implementation of the plant design, including the responsibilities and authorities of the design organization, implementation plan, and schedules. These requirements do not conflict with the Commission's regulations and guidance, and are, therefore, acceptable.

11.9 Configuration Management

Section 11.9 of Chapter 1 of the Evolutionary Requirements Document states that the plant designer should develop a configuration management program to be used throughout all phases of the plant's life, including the design phase. The program is intended to ensure that documents used in defining the design basis, technical baselines, as-built configuration, and procedures will be available over the life of the plant.

This section originally included information about the plant security system. In its letter dated October 9, 1991, EPRI elected to remove documents and information that would require protection as safeguards information from the lists of plant documentation specified in Attachments 1 and 2 to Section 11. The staff concludes that this deletion addresses its concern raised in its letter dated August 19, 1991, and is, therefore, acceptable.

The requirements in this section do not conflict with the Commission's regulations and guidance and are, therefore, acceptable.

11.10 Design Integration

In Sections 2.2.F.3 and 2.2.F.4 of the DSER for Chapter 1, the staff recommended that the use of "living" probabilistic risk assessment (PRA) models in developing the overall design be required in the Evolutionary Requirements Document. The staff's evaluation of Appendix A to Chapter 1, which discusses EPRI's proposed PRA guidelines, is provided in the corresponding appendix of this report and supersedes this issue. Therefore, this DSER confirmatory issue is closed.

Section 11.10 of Chapter 1 of the Evolutionary Requirements Document specifies requirements to ensure that the design of systems and subsystems of an ALWR will be acceptably integrated to minimize the need for redesign and backfitting, to minimize failure rates, and to minimize planned and unplanned outage times. EPRI specifies the use of modeling, plant simulation, documentation, and evaluation of operating experience to achieve such minimization. These requirements do not conflict with the Commission's regulations and guidance and are, therefore, acceptable.

11.11 Interdisciplinary Design Reviews

Section 11.11 of Chapter 1 of the Evolutionary Requirements Document requires that an interdisciplinary design review be conducted by a number of highly skilled groups of distinct disciplines, in order that participation in the design reviews by a wide range of experts will help to uncover potential problems during the design stage.

In its letter dated March 1, 1991, the staff stated that although experts in many disciplines were specifically mentioned as being required to participate in the review group, experts in physical security were not included. In its letter dated May 13, 1991, EPRI noted that the makeup of the design review team was limited to the major engineering disciplines and activities because it was impractical to list all possible specialties that will be required for the review. Nevertheless, in its June 11, 1991, markup, EPRI modified Section 11.11.2 to specify that additional special disciplines will be included as appropriate for the subject under review. The staff concludes that this change resolves its concern and is acceptable.

The requirements in this section do not conflict with the Commission's regulations and guidance and are acceptable.

11.12 Information Management System

Section 11.12 of Chapter 1 of the Evolutionary Requirements Document specifies that the plant designer will use appropriate computer hardware and software to establish, manage, and operate an information management system (IMS) during the design process and will provide the IMS to the plant owner for use during construction and operation. EPRI states that the objectives of the IMS are the following:

- provide an effective means to acquire, store, retrieve, and manipulate the documents and data necessary to design, construct, start up, operate, and maintain the plant
- make effective use of computer-aided design and engineering during design, construction, and operation
- provide for implementation of a PIN as discussed by the staff in Section 11.7 of this chapter
- use the computer in configuration management
- ensure that information needed for construction and operations is available

Attachment 1 to Section 11 of Chapter 1 gives the functional and operational requirements for the IMS.

In its letter dated May 13, 1991, EPRI committed to revise the computer and network security provisions of Section 5.9 of Attachment 1 to clarify that these provisions apply to protection from unauthorized disclosure of sensitive information as well as data loss or contamination. The staff has verified that these changes were included in Revision 3 of this section.

The staff discussed EPRI's proposed requirements for this system in SECY-91-226, "NRC Use of Computer and Computer Graphics-Aided Reviews for Advanced Reactor Designs and Nuclear Facilities," dated July 29, 1991. The industry is beginning to develop programs to define the requirements for information management systems for designing and operating new reactors and nuclear facilities. The staff is establishing an advanced technology group to develop strategies to ensure that the staff understands how the industry will use computer technologies to design, construct, and operate advanced reactors and nuclear facilities and to ensure that the NRC establishes an effective regulatory process to be consistent with the industry's approach to the use of these technologies.

11.13 Design/Construction Integration

Section 11.13 of Chapter 1 of the Evolutionary Requirements Document states that the construction schedule should be based on a standard plant design that is essentially complete, except for required site-specific engineering. The requirements of this section are intended to ensure that design documentation is completed on a schedule that minimizes its effect on the construction schedule.

Attachment 2 to Section 11 of Chapter 1 summarizes ALWR design activities. Item 1 in that attachment originally listed activities to be completed for design certification and safety determination. In its letter dated May 17, 1991, EPRI stated that because design certification and combined license rulemaking were beyond the scope of the Evolutionary Requirements Document, it would remove the reference to design certification. The staff has verified that this change was made in Revision 3 of this section and is acceptable. Item 2 in Attachment 2 summarizes in detail design engineering required to support the construction schedule. Some of the detailed design items in Item 2 have asterisks to indicate that parts of those items will be completed in the standard plant design, rather than by the plant owner. In its letter dated May 13, 1991, EPRI agreed to revise Item 2 to indicate that portions of the security plan (layout and detail drawings) will be part of the standard plant design. These changes were included in EPRI's June 11, 1991, markup; however, EPRI later elected to delete the security plan (layout and detail drawings) from Attachment 2 in response to the staff's concerns regarding protection of the information from unauthorized disclosure. The staff concludes that this is acceptable because it satisfies its concern regarding protection of safeguards information.

These requirements do not conflict with the Commission's regulations and guidance. However, as discussed by the staff in Section 7 of this chapter, the owner/builder must ensure that construction activities permit verification of the acceptability of the plant configuration in accordance with the requisite NRC Inspection Manual chapters. The staff will address this matter during its review of an individual application for a combined license.

11.14 Engineering Field Verification of As-Built Conditions

Section 11.14 of Chapter 1 of the Evolutionary Requirements Document states that the plant designer should identify and perform necessary engineering field verification activities to confirm the adequacy of the installation. The requirements in this section do not conflict with the Commission's regulations and guidance and are, therefore, acceptable.

11.15 Conclusion

The staff concludes that, with the exceptions noted above, the requirements in Section 11 of Chapter 1 of the Evolutionary Requirements Document do not conflict with the Commission's regulations and guidance and are, therefore, acceptable.

12 MECHANICAL EQUIPMENT DESIGN REQUIREMENTS

12.1 Introduction

Section 12 of Chapter 1 of the Evolutionary Requirements Document specifies requirements for the mechanical equipment used in many systems of a nuclear plant.

12.2 Inservice Testing of Pumps and Valves

In Section 2.2 of the DSER for Chapter 1, the staff stated that EPRI had agreed to include an appropriate reference to ASME Code, Section XI, in a future revision of Chapter 1. This reference has been acceptably added to Chapter 1. Therefore, this DSER confirmatory issue is closed. However, as a result of its review of the detailed information in Chapter 1 relative to inservice testing of pumps and valves, the staff requested that EPRI supply additional information, as discussed below. This discussion addresses issues identified by the staff in Chapters 5, 6, and 8 of this report.

Sections 12.2.7.1 and 12.4.3.2 of Chapter 1 of the Evolutionary Requirements Document state that system designs will include provisions for inservice testing of essential pumps and valves in accordance with American National Standards Institute/American Society of Mechanical Engineers (ANSI/ASME) OM-6, "Inservice Testing of Pumps," and OM-10, "Inservice Testing of Valves." These two standards are now referenced in ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."

In SECY 90-016, the staff concluded that the requirements of Section XI of the ASME Code provide information on the operational readiness of the components, but, in general, do not necessarily provide for the verification of the capability of the components to perform their intended safety functions. It concluded that the ASME Code does not ensure the level of component operability that is desired for the evolutionary ALWR designs (see the staff's regulatory departure analyses in Appendix B to this chapter).

Accordingly, in SECY-90-016, as supplemented by the staff's April 27, 1990, response to comments by the Advisory Committee on Reactor Safeguards (ACRS), the staff recommended criteria to the Commission to be used to supplement Section XI of the ASME Code. In its staff requirements memorandum on SECY-90-016, dated June 26, 1990, the Commission approved the staff's position for evolutionary LWRs. In addition, the staff agreed with the recommendation of the ACRS that the guidelines of Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," be applied to evolutionary LWR designs. The ACRS also recommended that the staff resolve the check valve testing and surveillance issue and that consideration be given to industry-proposed alternative ways of meeting inservice testing and surveillance requirements. The staff's review of the inservice testing programs for pumps and valves in both the evolutionary and passive ALWR designs is based on these guidelines.

In its letter dated December 22, 1989, EPRI responded to the staff's questions on Chapter 8 of the Evolutionary Requirements Document. In these responses, EPRI committed to modify Section 12 of Chapter 1 to address the staff's concerns. However, in Section 3.2 of the DSER dated January 15, 1991, for

Chapter 8 of the Evolutionary Requirements Document, the staff concluded that EPRI's responses were not entirely acceptable. To satisfactorily implement the enhanced criteria discussed above, the staff concluded that additional modifications should be made to the Evolutionary Requirements Document. In its letter dated May 17, 1991, the staff requested additional information on the Evolutionary Requirements Document. The staff noted in that letter that, with certain exceptions, the request for additional information (RAI) was applicable to both evolutionary and passive plants. EPRI responded to that RAI in its letter dated August 1, 1991. The evaluation that follows is based on the information submitted in Revision 3 of the Evolutionary Requirements Document and on EPRI's August 1, 1991, response.

12.2.1 Scope of the Inservice Testing Program

In its response to the staff's request to clarify the term "essential" in regard to the scope of the inservice testing (IST) program for pumps and valves, EPRI stated that for its ALWR program, an "essential" component or system is viewed by the utilities as important for safety, reliability, and availability and for protecting their investment. This response is consistent with the staff's position that all safety-related equipment, including non-Code class safety-related pumps and valves, should be tested in accordance with ASME Codes, Section XI, and is, therefore, acceptable.

Pump and Valve Reliability

In its letter dated May 17, 1991, the staff requested that EPRI provide a commitment to undertake a reliability assurance program to identify key components, to assess the applicability of existing data to these components, and to develop reliability estimates for these components. In its letter dated August 1, 1991, EPRI referred to several sections in Chapter 1 and one section in Chapter 5 of the Evolutionary Requirements Document. These sections generally require the plant designer to develop reliability estimates for key components and to perform sensitivity studies to better understand the impact of data uncertainties. This response is too broad to be completely acceptable. Specifically, with respect to existing data, the staff requires a commitment to assess the applicability of such data to safety-related components. Furthermore, where existing reliability data are not applicable, the expected reliability of those components will have to be developed through testing. Pending such a commitment, the staff will evaluate individual applications for FDA/DC in accordance with the above position. On the basis of the above discussion, this issue is closed.

In its letter dated May 17, 1991, the staff requested that the ALWR reliability program be used to determine actions necessary to improve component reliability and to maintain the desired component reliability level through the life of the plant. In its August 1, 1991, response, EPRI referred to several sections in Chapter 1 that were related to reliability and availability. In particular, it referred to Section 6.2.3 of Chapter 1, Revision 3, which states that the plant designer will prepare an analysis showing that the designs of the plant systems and supporting maintenance systems, recommended spare parts, surveillance tests, and test intervals will be adequate to meet the availability requirements. In the rationale portion of that section, EPRI stated that analyses relating plant system and component reliability to availability for the ALWR are necessary for making adequate decisions about plant system design. EPRI states that the sections referenced

in its August 1, 1991, response provide requirements to ensure that appropriate actions will be taken to improve reliability and that component reliability levels will be maintained. This response provides the requested information and is, therefore, acceptable.

In its letter dated May 17, 1991, the staff requested that testability of safety-related valves be provided for early in the design phase. The valve testability issue is discussed in Section 12.2.2 below.

12.2.2 Valves

During its review of Chapter 8 of the Evolutionary Requirements Document, the staff requested that EPRI propose a frequency of disassembly and inspection of safety-related check valves and motor-operated valves. EPRI stated that the frequency of valve inspections was beyond the scope of the Evolutionary Requirements Document. As discussed in Section 3.2 of the DSER for Chapter 8, the staff concluded that EPRI's response was not acceptable. In its May 17, 1991, letter, the staff again requested, as a minimum, a commitment to develop a program that will establish the frequency and extent of disassembly and inspection of safety-related valves, including the basis for the frequency and extent of each disassembly. In Revision 3 of Section 12.2.7.8 in Chapter 1 of the Evolutionary Requirements Document, EPRI provided a commitment that is consistent with the staff request.

In its letter dated August 1, 1991, responding to a concern raised on check valve testing requirements and testability, EPRI stated that the requirements in Section 12.2.7 of Chapter 1 of the Evolutionary Requirements Document were in compliance with the Commission's guidance in the staff requirements memorandum dated June 26, 1990, on SECY-90-016, and that a detailed design analysis considering such factors as component design, application, PRA insights, and design alternative was the appropriate method to determine the best valve for the application and the best testing method. EPRI stated further that for safety-related check valves that require testing, the plant designer will submit the details for NRC approval at the FDA/DC phase of each passive ALWR. The staff concludes that this response is not entirely acceptable. Although this issue is applicable to both the evolutionary and passive ALWRs, EPRI has not addressed it for the evolutionary plant design. The staff disagrees with EPRI's position that a commitment to check valve testing methods requires a detailed design analysis and concludes that a commitment to check valve testing methods should be part of the Evolutionary Requirements Document. In addition, a requirement should be added to the list of guidelines from EPRI NP-5479, "Application Guidelines for Check Valves in Nuclear Power Plants," in the requirement portion of Section 12.2.6.1 of Chapter 1 to state that, in the selection and application of valves, the plant designer should also consider parts clearance, disc stability, and wear relative to actual operational flow conditions. Pending such changes, the staff will review individual applications for FDA/DC in accordance with the above positions. On the basis of the above discussion, this issue is closed.

In its letter dated May 17, 1991, the staff requested that EPRI revise Section 12.2.7.2 in Chapter 1 to reflect the staff's position on full-flow testing of check valves as described in the letter. In its August 1, 1991, response, EPRI referred to its position as provided in its responses to RAI 210.39(b) and (f). For reasons similar to those discussed above, EPRI's response is not acceptable. The staff maintains that testing method and

testability are important parts of reliability assurance and that a commitment to the staff's position on full-flow testing of check valves should be part of the Evolutionary Requirements Document. Pending such a commitment, the staff will review individual applications for FDA/DC in accordance with the above position. On the basis of the above discussion, this issue is closed.

In its letter dated May 17, 1991, the staff requested additional information regarding the use of nonintrusive diagnostic techniques for check valves. In Revision 3 of Section 12.2.7.4 of Chapter 1 of the Evolutionary Requirements Document, EPRI states that check valve and system design will include provisions for the use of nonintrusive diagnostic techniques. This commitment is consistent with the staff's May 17, 1991, request and is, therefore, acceptable.

In its letter dated May 17, 1991, the staff requested additional information on motor-operated valve (MOV) testing requirements. Revision 3 of Sections 12.2.7.6 and 12.3.2.3.4 of Chapter 1 reflects the staff's comments relative to demonstrated design-basis capability. In particular, where in situ design-basis testing is not practical, MOVs with well-understood performance characteristics will be used. Also, requirements for preinstallation testing have been expanded to include full and partial design-basis conditions. Justification will be required to show that qualification tests apply to all other valves of the same type, size, and conditions. The staff concludes that the above commitments are consistent with staff positions relative to this issue and are, therefore, acceptable.

In its May 17, 1991, letter, the staff requested additional information on qualification testing of active and nonactive MOVs. In its August 1, 1991, response, EPRI stated that nonactive MOVs should be designed for potential mispositioning, but qualification testing was not required. This statement does not completely agree with the current staff guidelines on valve mispositioning. The staff concludes that nonactive MOVs in a safety-related system either should be designed to prevent mispositioning or should be required to be subjected to qualification testing to demonstrate their capability to recover from mispositioning. Mispositioning may occur through actions taken at any time locally (manual or electrical), at a motor control center or in the control room, and includes deliberate changes of valve position for performing surveillance testing. Therefore, the staff will review individual applications for FDA/DC in accordance with the above position. On the basis of the above discussion, this issue is closed.

Recent industry experience and the results of NRC inspections of MOV programs indicate several areas require attention in the EPRI document. Specifically, in addition to the technical information described in Section 12.2.2.5 of Chapter 1 of the Evolutionary Requirements Document that is to be provided with each valve, operator loads as a function of fluid temperature (subcooling) and seismic/dynamic effects, as well as precise internal dimensions of the valve, should be provided. In addition to consideration of stem leakage in establishing the proper globe valve orientation described in Section 12.2.2.6.2 of Chapter 1, any reliance on a globe valve to isolate flow or the use of the valve for throttling flow should also be considered in establishing proper orientation. In addition to ensuring that the valve bonnet and disc will be designed to prevent pressurization due to heatup of fluid trapped in the bonnet as described in Section 12.2.2.8.2 of Chapter 1, the bonnet should be designed so that its internal pressurization is not

greater than that of both the upstream and downstream piping, or the motor operator should be designed to overcome such pressurization. EPRI should revise Section 12.3.2.3.3 of Chapter 1 to require that provisions be made for measuring both stem thrust and actuator torque because of the importance of information regarding the conversion of torque to thrust (i.e., stem factor). As a clarification of Section 6.2.2.1.4 of Chapter 5 regarding the capability of isolation valves to close against conditions that may exist during events requiring containment isolation, the isolation valves should be designed and test-qualified to be able to isolate flow resulting from a pipe break at the worst-case differential pressure (e.g., a condition resulting from a failure to scram the reactor in a timely manner), because the potential for a break in a line from the reactor vessel would likely be greatest when the reactor pressure was abnormally high. Pending modification of these sections, the staff will review individual applications for FDA/DC in accordance with the above positions. On the basis of the above discussion, this issue is closed.

In its letter dated May 17, 1991, the staff requested that EPRI address compliance with General Design Criterion (GDC) 54 of Appendix A to 10 CFR Part 50 with respect to containment isolation valve (CIV) design and to commit to leak test these valves to appropriate limits. GDC 54 includes requirements to design for the capability to leak test CIVs in all piping systems penetrating the containment. This requirement applies to the primary and secondary systems penetrating the containment. In its letter dated August 1, 1991, EPRI referred to Table B.1-1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document, which states that the design will meet GDC 54.

In its May 17, 1991, letter, the staff also discussed some limitations of the Type C leak rate testing requirements of Appendix J to 10 CFR Part 50 for CIVs. In particular, Appendix J Type C leak rate testing adequately determines the containment leaktightness provided by the valves included in this test program. However, these tests do not require that individual valve leakage limits be defined, nor is corrective action required based on individual valve leakage rates. ASME/ANSI OM-10 only requires CIVs to be tested in accordance with Appendix J. Therefore, the staff requested that EPRI revise the Evolutionary Requirements Document to require analyses of leakage rates and corrective actions for CIVs. Acceptable requirements for the analysis of leakage rates and corrective action are contained in either Paragraph IWV 3421-3427(a) of Section XI of the ASME Code or Paragraph ISTC 4.3.3 of ASME/ANSI OM-10, 1990.

In its response dated August 1, 1991, EPRI explained that the requirement in Section 6.2.2.2 of Chapter 5 of the Evolutionary Requirements Document was intended to require the plant designer to minimize the number of valves that will be subjected to Type C testing in accordance with Appendix J to 10 CFR Part 50 rather than to set down the type of testing required for CIVs. EPRI's response also referred to Section 12.2.7.1 of Chapter 1 of the Evolutionary Requirements Document, which requires the plant designer to provide for testing of essential valves in accordance with ASME/ANSI OM-10. Furthermore, EPRI stated that the designation of specific inservice testing requirements was beyond the scope of the Evolutionary Requirements Document and properly belonged in FDA/DC documentation. The staff disagrees with EPRI's position that the designation of specific inservice testing requirements is beyond the scope of the Evolutionary Requirements Document. For the reasons discussed above, the staff has also determined that EPRI's response will not result in

individual CIV leak rate testing and is, therefore, not acceptable. The staff will review individual applications for FDA/DC in accordance with the above position.

12.2.3 Pumps

As stated in Section 3.2 of the DSER for Chapter 8 and the staff's letter dated May 17, 1991, it is the staff's position that the pumps for ALWR designs should be provided with instrumentation to verify that the net positive suction head (NPSH) is greater than or equal to the NPSH required during all modes of pump operation. In Section 12.4.3.2 of Chapter 1 of the Evolutionary Requirements Document, EPRI indicates that its provision to test pumps in accordance with ASME/ANSI OM-6 will ensure that sufficient instrumentation and test connections are provided to monitor pump performance and trend degradation. However, OM-6 does not address the above staff position. Therefore, the staff requires a more explicit commitment to provide instrumentation to measure suction pressure. Since EPRI has not responded to this staff concern, the staff will review individual applications for FDA/DC in accordance with the above position. On the basis of the above discussion, this issue is closed.

In its letters dated December 22, 1989, and August 1, 1991, EPRI responded to the staff requests for additional information regarding the capability for pump flow testing. EPRI committed to modify Section 12 of Chapter 1 in accordance with the staff position that system design should provide the capability for pump flow testing at rates at least as large as the design flow rate. However, in the revised Section 12.4.3.1 of Chapter 1, EPRI states that system design will provide the capability for pump flow testing on a periodic basis at a flow rate that adequately verifies operability and will not result in pump degradation. It further states that, if practicable, the capability for system flow testing during normal operation will be provided and that, as a minimum, the capability will be approximately 60 percent of the design flow rate. However, 60 percent of the design flow rate is not acceptable. Therefore, the staff will review individual applications for FDA/DC in accordance with its position that the capability to test at 100 percent of the design flow rate should be provided. On the basis of the above discussion, this issue is closed.

In response to the staff's concern regarding the adequacy of the mini-flow system as discussed in Section 3.2 of the DSER for Chapter 8, EPRI deleted the phrase "25 percent of the pump design flow" from the rationale portion of Section 12.4.1.3 of Chapter 1 of the Evolutionary Requirements Document. In addition, EPRI states that system designs will be such that the pumps will not operate below the minimum flow required for pump protection for all operating modes. If minimum recirculation flow lines are required, these lines will be sized to ensure that degradation will not occur as a result of continuous operation on bypass. The design of the minimum flow lines, if installed, will permit periodic testing to verify the flow is in accordance with design. This commitment addresses the staff's concern, conforms to the applicable staff guideline, and is, therefore, acceptable.

In Section 3.2 of the DSER for Chapter 8 and in its letter dated May 17, 1991, the staff requested that EPRI provide a commitment to periodically disassemble and inspect all safety-related pumps. The staff requires, as a minimum, a commitment to develop a program that will establish the frequency and the

extent of disassembly and inspection of safety-related pumps, including the basis for the frequency and extent of each disassembly. EPRI has not responded to this request for the evolutionary plant. Pending such a commitment, the staff will review individual applications for FDA/DC in accordance with the above position. On the basis of the above discussion, this issue is closed.

12.2.4 Conclusion

The staff concludes that, with the exceptions noted above, Section 12 of Chapter 1 of the Evolutionary Requirements Document satisfies the staff guidelines discussed in Section 12.2 of this chapter relative to inservice testing of pumps and valves and is, therefore, acceptable.

12.3 Radiation Protection Considerations

Pumps

Section 12.4 of Chapter 1 of the Evolutionary Requirements Document states that pumps will be designed with flanged connections to facilitate removal and replacement. Pumps in radiation areas will have long-life bearings and permanent-type lubrication where practical. Pumps in nuclear service will be provided with drain and flush connections to facilitate decontamination.

Tanks

Section 12.6 of Chapter 1 states that all tanks will be designed to prevent the unintended retention of particulate material by the incorporation of one or more of the following features: sloped or cone-shaped tank bottoms, grinding of internal welds to minimize crud traps, tank flushing capability, and lancing or chemical cleaning capability.

Filters and Ion Exchangers

Section 12.9 of Chapter 1 states that process vessel manways will be sized to accommodate personnel wearing anticontamination clothing and remote or semi-remote tools will be available to facilitate maintenance of vessel internals. Filters used will require minimal time and effort to keep in service. Cartridge filters that will have high radiation levels will be designed to be changed remotely. Standardized techniques for filter handling will be used throughout the plant to ensure that doses will be ALARA during filter maintenance and changeout. Bag filter housings will be designed for ease of removal. Such features are intended to minimize the personnel radiation exposure associated with changing filters and, therefore, are acceptable.

Ion exchangers will have downstream resin traps to prevent the escape of resins in the event of a failure of an internal screen in the ion exchanger. These resin traps will be located outside the ion exchanger enclosure (in a lower radiation area) and will be provided with a remote backwash capability. This remote backwash capability will permit the resin traps to be cleaned remotely from a low-dose-rate area, thereby eliminating the associated personnel radiation exposure normally accrued during cleaning of resin traps.

Conclusion

The staff concludes that the design features of Section 12 of Chapter 1 of the Evolutionary Requirements Document do not conflict with the applicable guidance in Regulatory Guide 8.8 for minimizing personnel exposures, and SRP Sections 12.1, "Assuring That Occupational Radiation Exposures Are as Low as Is Reasonably Achievable," and 12.3-12.4, "Radiation Protection Design Features," and, therefore, are acceptable.

13 CONCLUSION

Chapter 1 of the Evolutionary Requirements Document

On the basis of its review, subject to resolution of the identified outstanding issues, the staff concludes that the requirements in Chapter 1 of the Evolutionary Requirements Document do not conflict with current regulatory guidelines and are acceptable. However, Chapter 1 contains requirements that are so general in nature that it is difficult for the staff to determine whether their implementation will meet the Commission's regulations, guidance, and practices. The staff's review of the remainder of the document is discussed in detail in the chapters that follow, providing a clearer picture of the extent to which the Evolutionary Requirements Document complies with the Commission's regulations. The conclusions regarding the staff's review of the entire Evolutionary Requirements Document are provided below. Applicants referencing the Evolutionary Requirements Document will be required to demonstrate compliance with the additional guidance in the Standard Review Plan (NUREG-0800) or provide justification or alternative means of implementing the associated regulatory requirements.

Evolutionary Requirements Document

On the basis of its review, subject to resolution of the identified outstanding issues listed in Section 1.4 of each chapter of this report, the staff concludes that the requirements in the Evolutionary Requirements Document (Volume II) 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 evolutionary design will be adequate. Therefore, applicants referencing the Evolutionary Requirements Document will be required to demonstrate compliance with the additional guidance in the Standard Review Plan or provide justification or alternative means of implementing the associated regulatory requirements.

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 this chapter provides that analysis. The staff forwarded these issues to the Commission in SECY-90-016, SECY-91-078, and a draft Commission paper issued on February 27, 1992.

In its SRM dated June 26, 1990, and August 15, 1991, the Commission provided its decisions on SECY-90-016 and SECY-91-078, respectively.

After the staff issues the February 27, 1992, draft Commission paper in final form, the Commission will complete its review of the basis for the approach that the staff is proposing for those issues and, accordingly, may at some future point in the review determine that those issues involve policy questions that the Commission may wish to consider. Because the Commission has not reviewed the approaches to resolving these issues, they do not represent agency positions. In addition, certain technical issues still have to be resolved before the staff can complete its review.

Therefore, the staff concludes that the Evolutionary Requirements Document (Volume II) specifies requirements that, subject to the resolution of the identified open issues and vendor- and utility-specific items, if properly translated into a design and constructed and operated in accordance with the NRC regulations in force at the time the design is submitted, should result in a nuclear power plant that will 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. In addition to complying with existing regulations, such a facility would also be consistent with the Commission's policies on severe-accident protection.

CHAPTER 1, APPENDIX A, "PRA KEY ASSUMPTIONS AND GROUNDRULES"

1 INTRODUCTION

This appendix of the SER documents the NRC staff's review of Appendix A, "PRA Key Assumptions and Groundrules," to Chapter 1 of Evolutionary Requirements Document through Revision 3. Appendix A to Chapter 1 was prepared, under the project direction of EPRI and the ALWR Utility Steering Committee, by Duke Power Company; Jack R. Benjamin Associates; SAROS; TENERA, L.P.; and EPRI.

On June 30, 1989, EPRI submitted Appendix A to Chapter 1 for staff review. By letters dated April 10, July 13, and August 17 and 23, 1990, the NRC staff requested that EPRI supply additional information. EPRI provided the information in its response dated November 7, 1990.

On November 4, 1991, the staff issued its DSER for Appendix A to Chapter 1. On December 11, 1991, the staff and EPRI met with the Advisory Committee on Reactor Safeguards (ACRS) Subcommittee on Improved Light Water Reactors to discuss this appendix, the staff's corresponding DSER, the outstanding issues from the staff's review of the appendix, and EPRI's approach to resolving each issue.

On September 7, 1990, EPRI submitted Revision 1 of the Evolutionary Requirements Document. Revisions 2, 3, and 4 were docketed on April 26 and November 15, 1991, and April 17, 1992, respectively. This appendix documents the staff's review of Appendix A to Chapter 1 of the Evolutionary Requirements Document through Revision 3, as well as its review of EPRI's responses to the staff's DSER that were docketed as pen-and-ink changes to the Evolutionary Requirements Document by letter dated January 24, 1992. EPRI has incorporated these changes as Revision 4.

Use of the Probabilistic Risk Assessment in Future Designs

It is not the Commission's intent that probabilistic risk assessments (PRAs) for evolutionary plants merely be used to validate an already frozen design, but rather that the PRAs be used as a design tool to improve the design, enhance safety, and help provide valuable insights into specific plant vulnerabilities. Consistent with this philosophy, the Commission expects evolutionary plant vendors to make use of PRA insights to help in ensuring that designs have an appropriate balance of prevention and mitigation from a severe-accident standpoint and that the designs benefit from PRA safety insights. The staff intends that its reviews of evolutionary plant PRAs will not focus on checking only the quality of the PRA, but also how well the vendor or utility continues the use of the PRA tool to minimize the estimated core damage frequency and offsite consequences.

In its DSER for Appendix A to Chapter 1, the staff identified the use of PRA in design as an open issue. In response to the staff's DSER, EPRI modified the Evolutionary Requirements Document to indicate that the design-specific PRA will be integrated into the design process to enhance and improve the

design (including meeting EPRI's core damage frequency and public risk objectives). Although the EPRI response is generally acceptable, FDA/DC applicants must also submit a description of the following:

- how PRA insights specifically influenced the design process
- what design features, if any, were added to or removed from the design as a result of PRA insights
- how plant operating experience was factored into the design-specific PRA
- how it was determined (including criteria) if there were any vulnerabilities in the plant design for internal and external events
- how the PRA was used to develop an appropriate balance of prevention and mitigation

The staff concludes that EPRI has provided guidance to vendors and designers on the use of a PRA during the design process and, therefore, this open issue is closed.

FDA/DC certification of a design will be based in part on a PRA of that design. Since the validity of a PRA is highly dependent on the assumed reliability of systems, structures, and components, the staff has determined that the plant designer should provide a reliability assurance program (RAP) as part of the FDA/DC application. This issue is discussed in this report in Section 6 of Chapter 1 and under Item II.M of the February 27, 1992, draft Commission paper on policy issues continued in Annex C of Appendix B to Chapter 1.

There will be three phases of the PRA for ALWRs:

(1) The development of the design-specific PRA by the ALWR vendor for FDA/DC.

(2) The updating of the PRA (or the use of PRA techniques) at the combined license stage to include site-specific external events.

(3) The updating of the PRA after the combined license has been issued to include as-built information such as actual equipment capacities, control room details, and updated human factors analysis. This phase is expected to be completed before startup of the plant and to continue through the life of the plant with the PRA being maintained as part of the RAP.

The PRA will be a useful tool in developing appropriate plant maintenance activities. The NRC has recently promulgated the maintenance rule (10 CFR 50.65), which requires commercial power plant licensees to monitor the effectiveness of maintenance for safety-significant plant equipment in order to minimize the likelihood of failures and events caused by lack of effective maintenance. The rule, which will become effective in July 1996, requires licensees to monitor the performance or condition of certain systems, structures, and components (SSCs). Licensees are to compare monitoring results against licensee-established goals in a manner sufficient to provide reasonable assurance that those SSCs will be capable of performing their intended

functions. The goals are to be established commensurate with the safety significance of each SSC. Key assumptions in the plant-specific PRA and the PRA insights are to be considered with respect to how the assumptions and insights should affect plant maintenance. The rule encourages licensees to develop "living" PRAs.

The issue of a living PRA was identified as a confirmatory issue in the DSER for Chapter 1 because Appendix A had not yet been submitted when the staff prepared the DSER. The staff concludes that EPRI's key assumptions and groundrules provide guidance to vendors on the conduct of a PRA. However, the staff will have to review the issue of a living PRA during its combined license review and during plant operation. Therefore, this confirmatory issue is closed.

1.1 Review Criteria

Section 1 of Volume 1 of this report describes the approach and review criteria used by the staff during its review of Appendix A to Chapter 1 of the Evolutionary Requirements Document.

1.2 Scope and Structure of Appendix A to Chapter 1

Appendix A to Chapter 1 of the Evolutionary Requirements Document defines the ALWR Utility Steering Committee's overall requirements for conducting a PRA.

The key topics addressed in the staff's review of Appendix A to Chapter 1 include

- scope of the PRA
- initiating events
- success criteria
- uncertainty analysis
- plant modeling
- human reliability
- containment analysis
- reliability data
- reliability assurance program
- seismic analyses
- fire analyses
- internal and external flood analyses
- use of PRA in design

1.3 Policy Issues

During its review of Appendix A to Chapter 1 of the Evolutionary Requirements Document, the staff did not identify issues that involve policy questions for the technical areas discussed in this appendix, 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 Appendix A to Chapter 1 of the Evolutionary Requirements Document contained the following outstanding issues:

Open Issues

(1) use of PRA in design (1)

(2) events other than those initiated during full-power operation (1.6)

(3) consideration of reactivity accidents (1.7)

(4) definition of core damage (1.7)

(5) uncertainty treatment (1.9)

(6) form of the results (1.10)

(7) initiating events (2.2)

(8) truncation (2.5)

(9) nested solution process (2.5)

(10) mission time (2.10)

(11) use of EPRI-proposed failure data (2.11)

(12) instrumentation and controls reliability data (2.11)

(13) tornadoes and extreme winds (3.2)

(14) external flooding (3.2)

(15) internal fires (3.2)

(16) internal flooding (3.2)

(17) quantitative uncertainty analysis for seismic events (3.3)

(18) hazards analysis for ground response spectrum (3.3)

(19) in-plant sequence assessment (4.5)

(20) containment event analysis (4.6)

(21) details of uncertainty analysis (4.6)

(22) implementation of EPRI's public-safety requirements (5.1)

(23) analysis of systems and sequences (6.1)

(24) assessment of containment response (6.2)

(25) analysis of source terms (6.3)

(26) use of physically based source term (6.3)

(27) human reliability analysis considerations (7.1)

(28) function, task, timeline, and link analyses (7.3)

(29) performance shaping factors (7.3)

(30) evaluation tools for performance shaping factors (7.3)

(31) quantification methods for human reliability analysis (7.3)

(32) reliability data base for consideration of loss of offsite power (Annex A)

(33) failure rate for the main step-up transformer (Annex A)

(34) ALWR reference site data (Annex B)

Confirmatory Issue

(1) generic data sources (7.3)

The final disposition of each of these issues is discussed in detail in the appropriate section of this appendix, as indicated by the parenthetical notation following each issue. All issues identified in the DSER for Appendix A to Chapter 1 have been resolved.

1.5 Vendor- or Utility-Specific Items

The vendor- or utility-specific items, with references to appropriate sections of this appendix given in parentheses, are listed below. The designators in front of each issue provide a unique identifier for each issue. The letter "E" indicates that the issue applies to evolutionary plant designs. The first number designates the chapter (or appendix) in which it is identified. The letter "V" designates that it is a vendor- or utility-specific item. The final number is the sequential number assigned to it in the appendix.

E.1A.V-1 use of PRA in design
E.1A.V-2 modeling of a PRA (1.6)
E.1A.V-3 shutdown and low-power events (1.6)
E.1A.V-4 external events (1.6, 3.3, and 6.1)
E.1A.V-5 core damage frequency (1.7)
E.1A.V-6 uncertainty treatment (1.9 and 6.1)
E.1A.V-7 documentation of method of truncation of accident sequences (1.10 and 2.5)
E.1A.V-8 low-frequency accident initiators leading to core damage (2.2)
E.1A.V-9 mission time (2.10)
E.1A.V-10 failure rate for components (2.11)
E.1A.V-11 tornadoes and extreme winds (3.2)
E.1A.V-12 external river flooding (3.2)
E.1A.V-13 hurricanes and storm surges (3.2)
E.1A.V-14 tsunami (3.2)
E.1A.V-15 internal fires (3.2)
E.1A.V-16 site-specific external events
E.1A.V-17 internal flooding (3.2)
E.1A.V-18 seismic hazards analysis (3.3)
E.1A.V-19 core-damage-sequence binning (4.1)
E.1A.V-20 plant damage state definition (4.2)

E.1A.V-21 containment isolation assumptions and criteria (4.3)
E.1A.V-22 in-plant sequence assessment (4.5)
E.1A.V-23 containment event analysis (4.6)
E.1A.V-24 details of uncertainty analysis (4.6)
E.1A.V-25 source term definition (4.7)
E.1A.V-26 event tree binning (4.8)
E.1A.V-27 risk measures related to containment performance (4.8)
E.1A.V-28 use of mean values for characterization of risk results (5.1)
E.1A.V-29 assessment of risk measures (5.2)
E.1A.V-30 calculation of offsite consequences (5.2)
E.1A.V-31 importance analysis for input to reliability assurance program (6.1)
E.1A.V-32 assessment of containment response (6.2)
E.1A.V-33 source term (6.3)
E.1A.V-34 scope and objective of human reliability analysis HRA (7.1)
E.1A.V-35 process and criteria to confirm adequacy of human reliability analysis (HRA) (7.2)
E.1A.V-36 impact of advanced technologies on HRA (7.3)
E.1A.V-37 function, task, timeline, and link analyses (7.3)
E.1A.V-38 generic data sources (7.3)
E.1A.V-39 performance shaping factors and their evaluation tools (7.3)
E.1A.V-40 quantification methods for HRA (7.3)
E.1A.V-41 loss of offsite power frequency (Annex A)
E.1A.V-42 site data (Annex B)

1.6 Overall Scope and Methods of Appendix A to Chapter 1

Appendix A to Chapter 1 of the Evolutionary Requirements Document defines the ALWR Utility Steering Committee's requirements for the PRA to be performed for evolutionary ALWRs. It defines the purpose and scope of the PRA, identifies previously developed methods to be used, identifies new or improved methods to be used, and defines procedures to be used if existing procedures are incomplete or conflicting. Such a PRA is required to be performed to satisfy the regulatory requirements in 10 CFR Part 52. The staff concludes that the overall scope and methods described in Section 1.1 of Appendix A to Chapter 1 are acceptable, except as described below.

Scope of Structures, Systems, and Components Modeled in PRA

It is unclear whether the Evolutionary Requirements Document requires the PRA modeler to include all important plant equipment. As clarification, the staff requires that ALWR vendors submit a PRA that models all equipment (1) that the NRC requires evolutionary plants to have available to prevent or mitigate severe accidents and (2) that an operator is likely to attempt to use to prevent or mitigate any initiating event that may lead to core damage. This must include consideration of accident management measures or strategies for preventing and mitigating severe accidents and for establishing and maintaining long-term cooling and containment heat removal.

Modes of Operation Other Than Full Power

Revision 1 of Appendix A to Chapter 1 called for (at a minimum) a qualitative screening evaluation to be performed for events during modes other than full

power. In the DSER for Appendix A to Chapter 1, the staff concluded that a qualitative evaluation alone was not adequate and identified this as an open issue.

In its January 24, 1992, response to the DSER (which referenced EPRI's letter dated December 16, 1991, responding to the staff's request for additional information dated September 5, 1991), EPRI modified the Evolutionary Requirements Document to address shutdown risk in more detail. EPRI stated that the Evolutionary Requirements Document will evaluate "all operating conditions (including shutdown ones)." In addition, it will require that the plant-specific PRA performed for shutdown conditions evaluate core damage frequency (only Level 1 PRA required) with a simplified evaluation of release frequencies and magnitudes. The study will attempt to identify potential vulnerabilities so that evolutionary plant designers can remedy the most important ones. The staff concludes that the shutdown analysis guidelines proposed by EPRI are well thought out and generally acceptable with the following exceptions:

- Although it is necessary to develop assumptions that limit the scope of a shutdown PRA in order to make the analysis tractable, in reality it is impossible to predict all combinations and permutations of maintenance activities that might occur during outages. However, on the basis of inspections performed at operating plants, it is apparent that the risks associated with planned outages can be significantly reduced when plant personnel are aware of the risk potential of certain configurations. Use of the simplified shutdown PRA can help outage planners eliminate configurations that are particularly risk significant, sensitize shutdown planners to the need to minimize the time of exposure for risk-significant configurations that cannot be avoided, and provide the planners with safety insights that can be shared with those performing plant maintenance. Therefore, at each planned outage, combined license applicants should reevaluate the expected shutdown configurations using the simplified shutdown PRA to help in configuration control, maintenance, and outage scheduling.

- Potential initiating events should not be excluded from consideration on the basis that safety functions are available and the window of risk is of short duration. For example, these criteria would screen out mid-loop operation if taken literally, even though mid-loop operation has been shown to be a potentially important contributor to shutdown risk.

- When determining the list of shutdown initiators to be considered, the analysts must consider existing (NRC) information notices that pertain to shutdown events and must consider internal flooding and fires.

- When making a functional assessment of the plant's response to a shutdown event, not only the effectiveness and types of alarms expected to be actuated should be analyzed, but also instrumentation pertaining to such things as vessel level and reactor coolant system temperature and all instrumentation specific to decay heat removal system performance (e.g., pump amperage). Quantitative analysis of the operator's ability to respond to a shutdown initiator requires an evaluation of the instrumentation available.

- Consideration must be given to including insights from the shutdown PRA analysis in the accident management program guidance the vendors will develop for combined license applicants and in the technical specifications for modes other than full power.

Therefore, the staff concludes that the issue involving events other than those initiated during full-power operation is closed.

External Events

At the FDA/DC stage, ALWR vendors must address seismic events, tornadoes, fires, and internal flooding. At the combined license stage, site-specific external events such as river flooding must be addressed. If a bounding analysis has not been performed by the ALWR vendor, combined license applicants must provide a site-specific analysis, using PRA techniques, that evaluates these external events. From the staff's viewpoint, the importance of these analyses comes not from bottom-line numbers, but rather from insights into plant design robustness, potential severe-accident vulnerabilities, and areas in which reliability and/or maintenance are particularly safety significant. In Section 3 of this appendix, the staff provides additional details regarding external events.

1.7 Definition of Core Damage

Section 1.2 of Appendix A to Chapter 1 of the Evolutionary Requirements Document states that core damage from events involving loss of coolant inventory and/or loss of core heat removal will be assumed to have occurred f, and only if, both of the following have occurred:

- The collapsed level in the reactor has decreased so that active fuel in the core has been uncovered.

- A fuel cladding temperature of 2200 °F or higher is reached in any node of the core as defined in a best-estimate thermal-hydraulic calculation.

In its DSER for Appendix A to Chapter 1, the staff stated that EPRI's characterization of core damage was incomplete in that reactivity accidents were not considered and identified this as an open issue. In response to the open issue, EPRI added a criterion for defining core damage resulting from reactivity accidents. For reactivity accidents, core damage is assumed to occur if the axial peak, radial average energy exceeds 280 cal/g. The staff concludes that the definitions of core damage due to loss of coolant inventory and or loss of core heat removal and due to reactivity accidents are acceptable for purposes of the PRA; therefore, this DSER open issue is closed.

Section 1.2.2 of the original version of Appendix A to Chapter 1 stated that the plant design will be such that a realistic assessment of the mean core damage frequency will produce a best estimate no higher than 1E-5 event per reactor-year, including both internal and external events. In the DSER for Appendix A to Chapter 1, the staff noted that this requirement was inconsistent with the requirements of Chapter 1 that refer to a mean annual core damage frequency and identified this discrepancy as an open issue. In response to the DSER, EPRI modified Section 1.2.3 of Appendix A to require that the plant design be such that a realistic assessment of the core damage frequency will produce an estimate no higher than 1E-5 event per reactor-year

(including both internal and external events). The wording proposed here is still unclear, since there is no rigorous statistical definition of an "estimate" or "annual core damage frequency." These could be represented by the mean, median, mode, or some other parameter. In addition, point estimates do not permit investigation of the effect that uncertainties may have on the insights from a risk assessment.

Although the staff concludes that the goal of 1E-5 event per reactor-year is useful for designers and for potential customers of evolutionary plant designs, the Commission has stated that it does not intend to regulate advanced plants to a specific numerical core damage frequency goal. Although many measures of central tendency exist, the Commission has chosen "mean values" as the appropriate measure for safety goals and other Commission and staff risk targets. Therefore, when reporting core damage frequency and other risk results, "mean values" should be reported to the extent possible. In Section 1.9 of this appendix, the staff discusses further uncertainty analyses. ALWR vendors and combined license applicants may submit other measures of central tendency or statistical estimates, but the staff will use mean values (with their corresponding uncertainty distributions) for gaining insights and making decisions. Therefore, this DSER open issue is closed. The staff's further evaluation of EPRI's public safety goal is provided in Section 5.1 of this appendix.

1.8 Point-Estimate Quantification

Section 1.3 of Appendix A to Chapter 1 of the Evolutionary Requirements Document states that for each primary event input into the PRA model, a point estimate will be derived to represent that event in calculating the frequency of event sequences. Section 1.3 also states the mean value will be the point estimate used for this purpose. These mean values will be propagated through the PRA models, and point-estimate frequencies will be obtained for core-damage sequences and radionuclide release categories of interest.

1.9 Uncertainty Treatment

Internal Event Uncertainties

Revision 1 of Section 1.4 of Appendix A to Chapter 1 of the Evolutionary Requirements Document stated that a qualitative uncertainty analysis will be performed as part of the PRA, and that this analysis will, as a minimum, involve the identification and description of the potentially important sources of uncertainty, and an assessment of the significance of these uncertainties with respect to the results and conclusions of the PRA. The document also stated that where necessary, the qualitative analysis will be supplemented by quantitative evaluations of the sensitivity of the results to key uncertainty issues to aid in investigating the significance of these sources of uncertainty. However, it did not distinguish between the uncertainty and sensitivity analyses that would be performed for the Level 1 versus the Level 2 and 3 portions of the analysis, or for internally versus externally initiated events.

In the DSER for Appendix A to Chapter 1, the staff concluded that uncertainty analyses should be such that the staff has reasonable assurance that the PRA reflects variability in (1) the significance of key actions, events, and

phenomena for the plant design; (2) the effectiveness of the accident mitigation systems and potential design improvements; and (3) the estimates of risk measures related to public health and safety. This was identified as an open issue in the DSER.

In response to this DSER open issue, EPRI proposed modifications to Section 1.4 of Appendix A to Chapter 1 that provide further guidance on how uncertainty is to be considered in the PRA and that more clearly require that the qualitative uncertainty analysis be supplemented by a series of quantitative sensitivity studies. The revision also emphasizes the importance of analysts giving careful, systematic consideration to the sources of uncertainty that could be important, and the impact that each of these sources might have on the results.

The EPRI response represents an improvement in the Evolutionary Requirements Document but falls short of adequately addressing the full range of the staff's concerns. In particular, Appendix A still does not require a quantitative assessment of uncertainty or provide guidance on how uncertainties should be addressed for the different portions of the analysis (Levels 1, 2, and 3) and for internal versus external events. The staff recognizes the difficulty in performing uncertainty analyses and, in some areas such as Level 2, the limitations of such analyses. Nevertheless, given the inadequate state of knowledge of severe accidents and the considerable benefits of performing an uncertainty analysis (particularly for Level 1), the staff considers the present guidance on uncertainty analyses in the Evolutionary Requirements Document to be insufficient.

The staff requires that a full uncertainty analysis be performed for the Level 1 portion of the PRA, with uncertainties propagated from basic events, including initiating event frequencies, data, common cause/mode failure, success criteria, and human errors. The Level 1 uncertainty analysis is to be performed in such a manner that the most important uncertainties that contribute to core damage frequency are determined. Uncertainty analyses should be such that the staff has reasonable assurance that the PRA reflects variability in (1) the significance of key actions, events, assumptions (such as those related to success criteria), and phenomena for the plant design and (2) the effectiveness of the accident prevention systems and potential design improvements.

With regard to Level 2 uncertainties, the staff agrees with the rationale portion of Section 1.4 of Appendix A to Chapter 1 that many of the most important sources of uncertainty do not readily lend themselves to meaningful quantitative treatment and may be similar to what have been labeled "modeling" uncertainties. Furthermore, the staff recognizes that the evolutionary plant designs will incorporate features intended to minimize or eliminate the challenges posed by certain severe-accident phenomena, and that such features will reduce the significance of uncertainties in related severe-accident phenomena by reducing the frequency of occurrence and/or magnitude of the challenge. Examples of such features include reactor depressurization systems to reduce the frequency of reactor vessel failure at high pressure, inerting of the containment to minimize the potential for hydrogen combustion, and incorporation of reactor cavity flooding systems to minimize concerns regarding core-concrete interactions. Accordingly, a full, quantitative uncertainty analysis will not be required for the Level 2 portion of the PRA. Rather, the staff requires (1) the implementation of a systematic process for identifying

issues and phenomena of greatest risk significance for the advanced designs and (2) a more thorough treatment of those issues and phenomena and their associated uncertainties as part of the Level 2 analysis, for example, in the containment event trees (CETs).

The objectives of the Level 2 uncertainty treatment are to acknowledge and represent, within the context of the containment analysis, the full range of outcomes for those issues that are highly uncertain. This is in contrast to a more simplified approach where uncertainty issues (perhaps bimodal in nature) are represented in the CET by a single, "best-estimate" outcome. It should be noted that this treatment of uncertainty is distinctly different from the type of uncertainty analysis described in NUREG-1150 ("Severe Accident Rates, An Assessment for Five U.S. Nuclear Power Plants") in which probability distributions are developed for key branch points and propagated through the CET to obtain statistical information such as 5 and 95 percent confidence limits.

If an ALWR vendor does not choose to perform a NUREG-1150-type Level 2 uncertainty analysis, the vendor's process for addressing Level 2 uncertainties should contain the following three major elements: (1) initial screening of issues for applicability to the ALWR design, (2) sensitivity analyses to further delineate issues of greatest risk significance, and (3) systematic analysis of issue uncertainty as part of the Level 2 analysis. The staff considers the EPRI guidance, which calls for a qualitative uncertainty analysis and supplementary quantitative sensitivity analyses, to be an acceptable starting point for this analysis. However, the evaluation called for by the Evolutionary Requirements Document must be structured in such a way as to identify and develop insights on key issues and phenomena, and then augmented by a more detailed treatment of the risk-significant issues in the quantitative analysis.

The staff will address the acceptability of the Level 2 uncertainty treatment as part of its review of an application for FDA/DC. The adequacy of the analysis will be judged on the basis of the completeness of issues considered, reasonableness of parameter ranges and screening criteria for sensitivity analyses, and the methodology for decomposing and propagating the range of potential issue outcomes in the analysis. The treatment of uncertainties for Level 2 must provide the staff reasonable assurance that the PRA reflects the significance of key actions, events, and phenomena for the plant design, as well as the effectiveness of the accident mitigation features.

Therefore, the staff concludes that the DSER open issue concerning uncertainty treatment has been resolved. It discusses this issue in greater detail in Sections 4, 5, 6, and 7 of this appendix.

External Event Uncertainties

External event uncertainty analysis requirements are discussed by the staff in Section 6.1 of this appendix. Either a limited sensitivity or a limited uncertainty analysis will be performed for all external events analyzed using PRA techniques. The staff does not expect ALWR vendors or combined license applicants to perform detailed and complete uncertainty analyses for external events.

1.10 Form of Results

In the DSER for Appendix A to Chapter 1, the staff identified an open issue concerning documentation of a PRA. The staff concluded that the documentation of the PRA must be adequate for its review. The Evolutionary Requirements Document should provide an analyst with guidance that ensures that the PRA documentation provides a traceable path and that enough information is presented so that a competent PRA analysis team can duplicate the results. The staff stated that it did not expect that a review would duplicate the analysis in all details. Review of the PRA could be facilitated by including, in addition to the written material, material on magnetic media, such as the fault trees, event trees, data, and dominant cutsets. In response to the DSER open issue, EPRI's guidance in Section 1.5 of Appendix A on presenting the results of the PRA has been significantly expanded, including models and quantification. The staff concludes that the guidance in the revised Evolutionary Requirements Document is adequate. Therefore, this open issue is closed. However, in addition to complying with the expanded guidance on documentation in Section 1.5, vendors must submit a description of the method of truncation used in the quantification process.

2 PLANT MODELING

2.1 Model Structure

Section 2.1 of Appendix A to Chapter 1 of the Evolutionary Requirements Document states that the plant will be modeled in terms of a set of initiating events, event consequences composed of function or system success or failure states, and logic models that describe combinations of basic events that define the possible success and failure states. Containment functions whose failure could influence whether or not core damage occurred and/or the frequency of core damage will be included in the event trees. Each end state of each event tree will be designated either "success" or "core damage." The core-damage sequences, when combined with success or failure of systems needed to preserve containment integrity, will be categorized and grouped into plant-damage states for downstream modeling of the containment processes. The staff finds the proposed modeling structure to be adequate.

2.2 Initiating Events

Section 2.2 of Appendix A to Chapter 1 of the Evolutionary Requirements Document states that the analyst will develop a comprehensive list of potential initiating events for consideration in the PRA. EPRI states that the analyst should consider input from summaries of operating experience for current-generation plants, PRAs for plants with similar design characteristics, and review of the system designs. In the DSER for Appendix A to Chapter 1, the staff stated that EPRI did not indicate that initiating events that degrade mitigating or core-damage-prevention systems should be comprehensively identified and realistically quantified. An example of such an initiating event is failure of a dc bus, which may result in a reactor transient and, at the same time, fail one train of one or several safety systems. This was identified as an open issue in the DSER.

In response to the staff's DSER, EPRI modified Section 2.2 by providing guidance so that analysts will review each ALWR plant system to identify any failure that could both initiate a plant trip and degrade one or more systems that would otherwise be available to prevent core damage or mitigate consequences. The staff concludes that this revision is acceptable and this open issue is closed.

However, in addition, the staff will review applications for FDA/DC to ensure that ALWR vendors consider (1) low-frequency accident initiators (usually screened out in conventional PRAs) leading to core damage that could significantly challenge prevention and/or mitigation equipment and (2) low-frequency events with very high consequences. The search for these new initiating events must be conducted in such a way that events that are capable of affecting plant safety can be identified even at frequencies about a few percent of the core damage frequency. Since it is expected that evolutionary plant designs will systematically reduce the expected frequency of "traditional" initiating events that lead to core damage, it is important to look for initiators that have been ignored previously by analysts or were considered to have been bounded by "traditional" initiators. Two examples illustrate initiators not normally considered in PRAs but which may be important for ALWRs: (1) multiple initiators (e.g., a loss of service water followed by an independent loss of offsite power before service water has been

recovered) and (2) low-frequency loss-of-coolant accidents (LOCAs) (e.g., multiple steam generator tube ruptures). The process used to make this search for initiators should be submitted.

2.3 Success Criteria

In Section 2.3 of Appendix A to Chapter 1 of the Evolutionary Requirements Document, EPRI requires that a definition of success and failure for each function be provided based on realistic analysis of plant response. If conservative data are used to economize resources, the analysts are cautioned to identify conservative assumptions and to review the results to ensure that use of conservatisms does not obscure insights from the results. The analysts must also exercise care to ensure that any assumption or criterion considered to be conservative in one context does not introduce a nonconservatism in some other area (e.g., it is conservative to assume that containment pressure is high following a LOCA for containment analyses, but it is a nonconservative assumption for emergency core cooling system pump runout calculations). The staff agrees that these measures are prudent and believes the basis for these measures given by EPRI is valid.

2.4 Sequence Logical Identity

Section 2.4 of Appendix A to Chapter 1 of the Evolutionary Requirements Document states that the plant model and the solution and quantification techniques will retain the logical identity of the basic events that constitute each sequence. It is necessary to specifically identify which basic-event combinations contribute to the frequency of the dominant event sequences in order to understand and check the realism of the results. EPRI does not consider it sufficient to calculate sequence frequencies only and states that the specific equipment conditions must be known in order to determine whether recovery by the operations staff is possible and to judge how likely such recovery may be. This approach is logical and defensible. The NUREG-1150 studies, as well as some earlier studies, showed that recovery actions must be identified and quantified at the cutset level; this is in essential agreement with the EPRI approach.

2.5 Quantification

Truncation

Section 2.5.1 of Appendix A to Chapter 1 of the Evolutionary Requirements Document discusses the truncation of accident sequence cutsets. The staff agrees with the statements in this section calling for retaining information regarding low-frequency sequences. This may be important for identifying those sequences with a relatively higher potential for containment failure, as well as for preserving the ability to subsequently assess the effects of certain risk-sensitive areas. In the DSER for Appendix A to Chapter 1, however, the staff stated that the guidance did not provide assurance that a large number of cutsets, each of which is a small contributor but which in their aggregate form an appreciable contribution, would not be eliminated from further consideration. The staff further stated that there was no mention in Section 2.5.1 of whether the truncation would be performed before or after recovery actions (which are cutset dependent in many cases) are accounted for.

In response to the staff's DSER, EPRI modified the requirements of Section 2.5 to provide assurance that large numbers of cutsets, each of which is a small contributor but which in their aggregate form an appreciable contribution, are not eliminated from further consideration. Section 2.5.1 was modified to state explicitly that truncation will be performed after recovery actions (which are cutset dependent in many cases) are accounted for. The staff concludes that this revision addresses its concern and is, therefore, acceptable. This open issue is closed.

Nested Solution Process

Section 2.5.2 of Appendix A to Chapter 1 discusses the nested solution process, which is a method of truncating certain cutsets of support systems before the support system is incorporated into the fault trees of systems depending on the support system. In the DSER for Appendix A to Chapter 1, the staff stated that the guidance given by EPRI was not sufficiently complete to determine whether the process was a source of significant error. The staff requires assurance that the truncation technique used does not result in underestimating common-mode failures between trains of the support system or between different systems. Also, the truncated cutsets must be demonstrated to be negligible, not only individually, but in their aggregate, even after recovery actions are taken into account on the dominant cutsets, to ensure that important insights are not lost.

In response to the staff's DSER, EPRI modified the requirements of Section 2.5.2 to ensure that truncation limits at intermediate steps in the quantification process are selected to ensure that sufficient event combinations are retained, consistent with the truncation guidance in Section 2.5.1. This is acceptable to the staff; therefore, this open issue is closed.

In addition, the staff will review applications for FDA/DC to ensure that individual PRAs carefully document the method of truncation of the accident sequences, so that the method can be evaluated.

2.6 Modeling of Dependencies

Section 2.6 of Appendix A to Chapter 1 of the Evolutionary Requirements Document requires that the potential for dependent failures be considered comprehensively and treated quantitatively using the best available methods. EPRI lists the following five types of dependencies that it requires be treated explicitly:

- sequence functional dependencies
- intersystem dependencies
- intercomponent dependencies
- dependencies due to human actions
- dependencies between core-cooling systems and containment systems

EPRI recognizes that dependencies have the potential to defeat redundancy in the design and, therefore, deserve careful attention in PRA. Because of the greater degree of redundancy in the design requirements for an ALWR, it is particularly important to understand the potential effects of dependencies on an integrated level for the plant. The staff agrees that the proper modeling of dependencies is essential when conducting any PRA.

2.7 Interaction and Modeling of the Containment Systems

Section 2.7 of Appendix A to Chapter 1 of the Evolutionary Requirements Document requires that the delineation of core-damage sequences be coordinated with the assessment of containment response to ensure that any effects of containment phenomena on the availability of the systems needed to prevent core damage are appropriately reflected in the event trees. The staff agrees with EPRI that these steps are necessary in the performance of the PRA to avoid a possible source of error.

2.8 Common-Cause Failures

Section 2.8 of Appendix A to Chapter 1 of the Evolutionary Requirements Document requires that direct component-to-component and system-to-system functional dependencies be specifically addressed in the plant model. This section applies to root-caused events leading directly to multiple component outages from the shared cause. Common-cause initiating events are explicitly addressed under external events and specific internal events in Sections 3 and 4 of Appendix A to Chapter 1 of the Evolutionary Requirements Document. The staff agrees that these dependencies must be included in the plant model if a realistic estimate of the core-damage sequence frequencies is to be obtained, and that neglect of these dependencies leads to an underestimate of the core damage frequency.

2.9 Human Interaction

Section 2.9 of Appendix A to Chapter 1 of the Evolutionary Requirements Document provides guidance on modeling human interactions in the PRA. The staff's evaluation of this aspect of the EPRI document is given in Section 7 of this appendix.

2.10 Mission Time

Section 2.10 of Appendix A to Chapter 1 of the Evolutionary Requirements Document states that a mission time of 24 hours will be used for equipment required to remain operable for successful core cooling after an initiating event. Recent PRA studies by the French indicate that long-term (up to 1 year) reliability of post-LOCA core cooling using recirculation from the containment sump can contribute appreciably to core melt frequency. On the basis of their findings that post-LOCA decay heat removal through emergency core cooling system operation can extend over a period of months, the French have identified system modifications and developed contingency procedures to improve the long-term reliability of recirculation cooling. Although the risk significance of long-term core cooling may not be as great for ALWR designs in absolute terms, the French studies suggest that greater attention should be paid in the PRA to events outside the 24-hour time window traditionally assumed in PRA. In the DSER for Appendix A to Chapter 1, the staff concluded that mission times greater than 24 hours should be quantitatively considered for key systems required for long-term cooling. This should include consideration of accident management measures or strategies for establishing and maintaining long-term cooling. A quantitative bounding analysis may be adequate and may eliminate the need to consider in detail mission times greater than 24 hours for transients. This was identified as an open issue in the DSER.

Although EPRI made minor modifications to Section 2.10 in response to the staff's DSER, the staff continues to consider the EPRI guidance on mission time to be inadequate. Therefore, on the basis of the considerations discussed above, as well as in Section 1.6 of this appendix, the staff concludes that ALWR vendors must include the following in their PRAs:

(1) The scope of the PRAs performed for ALWR designs must be extended to include treatment of the plant evolutions and system functions necessary to bring the reactor to cold shutdown and to maintain this condition for the long term.

(2) Mission times are to be determined and justified by the ALWR plant vendor consistent with this expanded scope. This will necessitate consideration of mission times considerably longer than 24 hours, as well as explicit treatment of actions by onsite operating staff and offsite support organizations (i.e., all recovery actions) which may need to be accomplished within this timeframe.

Determination of an appropriate mission time should be based on consideration of system performance and reliability late in an event, availability and reliability of available backup systems and components, actions required to be taken by operating staff and offsite response organizations, and provisions that would be in place to ensure such actions could be taken in a timely manner. Mission times and success criteria will have to be justified by the plant designer and will be evaluated by the staff when it reviews the ALWR design-specific PRA as part of its review of an application for FDA/DC. This DSER open issue is closed.

2.11 Reliability Data

Section 2.11 of Appendix A to Chapter 1 of the Evolutionary Requirements Document discusses reliability data and suggests the use of the data base given in Annex A of Appendix A. In the DSER for Appendix A to Chapter 1, the staff concluded that Annex A of Appendix A to Chapter 1 should be modified to state that the analyst will have to review the failure data for applicability to each particular design, considering the operational environment of the equipment and the procedures applicable to the equipment. This applies to all the reliability data, including common-cause-failure parameters. This was identified as an open issue.

In response to the staff's DSER, EPRI modified Annex A of Appendix A of Chapter 1 as stated above. The staff finds this acceptable; therefore, this DSER open issue is closed.

In the DSER, the staff also stated that Annex A of Appendix A should be expanded to include data for systems anticipated in the ALWRs such as multiplexors, microprocessors, and cathode ray tube displays and identified this as an open issue.

In response to the staff's DSER, EPRI stated in a letter dated January 24, 1992, that generic data for the types of components cited were not widely available. In particular, failure rates for microprocessors varied substantially, depending on the complexity of the device and the specific application. In these cases, EPRI proposed to leave it to the analyst to identify the most applicable source of data. The staff concludes that EPRI's response

is acceptable and this open issue is closed. The staff will review FDA/DC applications to ensure that vendors provide detailed failure rate data for components that perform the essential functions of instrumentation and control systems.

At the combined license stage, the staff will review the site-specific loss-of-offsite-power frequency and estimates of the site-specific probability of recovery of offsite power as a function of time after its loss. Loss-of-offsite-power frequency and failure rates of high-voltage transformers are discussed by the staff in Annex A of this appendix.

The staff requires that test and maintenance intervals be considered when estimating the reliability of equipment. However, it may not be possible to test the full or integrated functions of some evolutionary design features. Some functions will never be tested in the as-built systems for reasons such as that their testing would result in the emptying of primary coolant into the containment. In some cases, safety-significant equipment will have long test intervals. As an example, since failures of motor-operated valves (MOVs) are well represented by a standby stress model, any MOV that has a long test interval is expected to have a high estimated failure probability. The testability of individual components is therefore an issue in addition to the testability of the whole function. On the other hand, testing at power of some components may lead to significant challenges to safety systems if additional random failures or human errors should occur.

Reliability data for non-safety-grade (but normally safety-grade in traditional PWRs and BWRs) components must be justified on the basis of the quality of the equipment purchased, test intervals, capability to perform its intended function in an adverse environment, experimental data, and applicable technical specifications.

3 EXTERNAL EVENTS

3.1 Identification of Initiating Events

Section 3.1 of Appendix A to Chapter 1 of the Evolutionary Requirements Document requires that the list of potential external initiating events taken from American National Standards Institute/American Nuclear Society (ANSI/ANS) 2.12, "American Nuclear Society Guidelines for Combining Natural and Man-Made Hazards at Power Reactor Sites," be considered for an ALWR PRA. EPRI has selected a single source for methodology to ensure consistent treatment of external events in PRAs.

3.2 Events That May Be Excluded Based on Qualitative Evaluation

EPRI has proposed that the following external events be excluded from a quantitative PRA analysis.

Tornadoes and Extreme Winds

In Section 3.2.3 of Appendix A to Chapter 1 of the Evolutionary Requirements Document, EPRI states that, except for loss of offsite power, tornadoes and extreme winds may be excluded on the basis of a qualitative evaluation. The staff disagreed and identified this issue as open in its DSER for Appendix A to Chapter 1. In response to the staff's DSER, EPRI modified its guidance for evaluating tornadoes. The new guidance in Section 3.3.1 of Appendix A to Chapter 1 requires ALWR vendors to provide a simplified quantitative assessment of tornadoes. This assessment is to take into account the potential for a tornado to cause an extended loss (at least 24 hours) of non-safety-grade systems that could play a role in core cooling. In addition, the analysis will consider independent random failures of safety equipment if the failure on demand is 1E-3 or greater. The staff concludes that EPRI's proposed approach is acceptable and this DSER open issue is closed. The staff will review the site-specific hazard data at the combined stage. As this issue is addressed in the staff's February 27, 1992, draft policy paper, the staff's final position is contingent on approval by the Commission.

External River Flooding

Section 3.2.17 of Appendix A to Chapter 1 states that external river flooding may be excluded on the basis of a qualitative evaluation. The staff disagreed and identified this issue as open in its DSER for Appendix A to Chapter 1. Subsequent submittals by EPRI were not sufficient to resolve the staff's concern. It is recognized that there are large uncertainties in the estimate of the river flood hazard, whether one is attempting to extrapolate from streamflow records or use runoff modeling methods. The National Research Council, in its publication, "Estimated Probabilities of Extreme Winds," dated 1988, states that "any method for estimating probabilities of floods greater than about the 100-year flood must include some form of extrapolation, a process that can, at best, introduce errors and, at worst, strain credulity." Therefore, the staff will review applications for a combined license to ensure that, at a minimum, the potential for large river floods is addressed explicitly in a bounding risk assessment. The staff will also accept a realistic assessment, using probabilistic techniques, of the chance of river flooding causing core damage at a particular site. Therefore, this DSER open issue is

closed. As this issue is addressed in the staff's February 27, 1992, draft policy paper, the staff's final position is contingent on approval by the Commission.

Hurricanes and Storm Surges

At some coastal sites, it is possible that very large hurricanes could generate winds and/or storm surges that would damage sufficient plant equipment to lead to core damage. As a minimum, the potential for hurricanes and storm surges causing core damage at coastal sites will have to be analyzed in a bounding risk assessment. It is also acceptable to submit a realistic probabilistic risk analysis of hurricanes and storm surges. The analysis submitted will be evaluated by the staff at the combined license stage.

Tsunami

For some coastal sites, it is possible that tsunamis could generate waves of sufficient height that they would damage sufficient plant equipment to lead to core damage. As a minimum, the potential for tsunamis causing core damage at coastal sites must be analyzed in a bounding risk assessment. It is also acceptable to submit a realistic probabilistic risk analysis of tsunamis. The analysis submitted will be evaluated by the staff at the combined license stage.

Internal Fire

Section 3.2.19 of Appendix A to Chapter 1 states that internal fires may be excluded on the basis of a qualitative evaluation. The staff disagreed and identified this issue as open in its DSER for Appendix A to Chapter 1. In response to the DSER, EPRI revised the Evolutionary Requirements Document to state that except for the main control room, a qualitative evaluation of severe-accident internal fires can exclude areas from consideration on the basis of a qualitative argument. The level of detail of a quantitative analysis of fires in the main control room would be a function of the estimated risk, based on first performing at least a bounding analysis. The staff concludes that the EPRI-proposed requirements in Sections 3.2.19 and 3.3.3 are still unacceptable. Therefore, it will review applications for FDA/DC to ensure that ALWR vendors perform either (1) a severe-accident fire analysis using the Fire- Induced Vulnerability Experiment methodology or (2) a fire PRA. These analyses are to be performed to gain insights into the adequacy of the design rather than as an exercise in determining the absolute value of estimated core damage frequency from internal fires. When performing the severe-accident fire analysis, ALWR vendors must address the following areas:

- Any fire area pinch point where multiple trains or divisions (including power or control cabling) are located in the same fire area must receive a quantitative assessment similar to that for the control room. In the fire analysis, the alternative shutdown capability must be analyzed and quantified from a human factors, systems function, and reliability standpoint.

- The fire analysis must evaluate whether safe shutdown can be achieved (and with what conditional probability) assuming that all equipment in any one fire area will be rendered inoperable by fire and that re-entry into the fire area for repairs and operator actions is not possible.

Because of its physical configuration, the control room is excluded from this approach, provided (1) an independent alternative shutdown capability that is physically and electrically independent of the control room is included in the design and is reliably capable of shutting down the reactor and (2) the likelihood of hot shorts involving power and control wiring associated with the remote shutdown panel function is negligibly small.

- The fire analysis must evaluate the probability that given a fire, fire protection for redundant shutdown systems in the reactor containment building will ensure that at least one shutdown division will be free of fire damage.

- The fire analysis must evaluate the likelihood that smoke, hot gases, or fire suppressant will migrate into other fire areas to the extent that they could adversely affect safe shutdown capabilities, including operator actions.

- The underlying assumptions used in the fire analysis (especially those related to barriers between divisions) are to be documented and submitted to the NRC staff.

- The fire analysis must evaluate barrier penetration reliability, because barriers can be compromised or be deficient.

- In a fire analysis, spatial separation in accordance with the requirements of Appendix R to 10 CFR Part 50 will not be considered adequate to prevent damage to equipment in the same fire area.

The staff will confirm the validity of the assumptions in the fire portion of the PRA during its review of a specific application for FDA/DC and again when the plant is in its final stages of construction. Therefore, this DSER open issue is closed. As this issue is addressed in the staff's February 27, 1992, draft policy paper, the staff's final position is contingent on approval by the Commission.

Other Site-Specific External Events

For a particular site, there may be specific external events that warrant severe-accident analysis such as accidents at nearby industrial or chemical plants, vulcanism, and transportation accidents. On a judicious basis at the combined license stage, applicants must address applicable site-specific external events either on the basis of a bounding risk analysis or a realistic probabilistic risk analysis.

Internal Flooding

Section 3.2.20 of Appendix A to Chapter 1 states that internal flooding may be excluded on the basis of a qualitative evaluation. The staff concluded that this was unacceptable and identified this issue as open in its DSER for Appendix A to Chapter 1. Subsequent submittals by EPRI were not sufficient to resolve the staff's concern. Even if the ALWR is better designed to prevent the initiation and spread of internal flooding than present-day plants, internal flooding could still contribute an appreciable amount to the estimated core damage frequency. Therefore, the staff will review applications

for FDA/DC to ensure that ALWR vendors provide either a systematic PRA of internal flooding or some simplified risk assessment that is capable of identifying vulnerabilities to internal floods in the design. However, the staff is unaware of any acceptable simplified method for evaluating internal floods other than that performed in the Yankee Rowe individual plant examination. This DSER open issue is closed. As this issue is addressed in the staff's February 27, 1992, draft policy paper, the staff's final position is contingent on approval by the Commission.

3.3 Events That Will Require Quantitative Assessment for Each ALWR

Seismic Events

The seismic design certification is based on a free field vibratory ground motion characterized by a response spectrum with a peak ground acceleration of 0.3g as described in Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants." At the combined license stage, the applicant is required to demonstrate that the site parameters are enveloped by the design certification values. If the site parameters exceed the design certification parameters, the combined license applicant will have to show sufficient seismic margin in design to meet the site-specific demand, in conformance with Section 2.5.2, "Vibratory Ground Motion," of the SRP.

Section 3.3.2 of Appendix A to Chapter 1 of the Evolutionary Requirements Document states that a seismic risk analysis must be performed as part of the PRA. It is the staff's position that a seismic PRA is not required.

The severe accident policy statement calls for the completion of a PRA and the consideration of severe-accident vulnerabilities exposed by the PRA in order to add to the assurance of public health and safety. The policy statement calls for the staff to review evolutionary designs for the purposes of concluding that the design is acceptable with regard to safety; it stresses the use of deterministic engineering analysis and judgment complemented by PRA. In addition, the policy statement calls for evolutionary plant vendors to use the PRA as a tool to consider a range of alternatives and combination of alternatives that address unresolved and generic issues and to search for cost-effective reductions in the risk from severe accidents.

In response to the policy statement, existing plant licensees were given the choice of using either a probabilistic risk or a margins methodology to search for potential vulnerabilities in the capability of the facility to deal with seismic events. A margins approach can provide a measure of the capability of the plant to withstand seismic events beyond the design basis or for some other specified external event.

Seismic PRAs and seismic margins assessments both have limitations and offer different insights about the capability of a plant to withstand severe accidents initiated by seismic events. PRAs can provide estimates of the frequency of severe accidents due to seismic events and can provide insights about the capability of the plant to survive seismic events. Two major advantages of PRA are that the response of all important systems is included in the analysis and failures in addition to seismic failures can be considered. The major disadvantage of a seismic PRA for a particular site is large uncertainties in the seismic hazard. The uncertainties would be even greater for a generic site. In light of these large uncertainties in seismic hazard,

the staff concludes that a seismic PRA for design certification should not be required. However, it believes insights about the capability of the plant to withstand a seismic event can be obtained from a margins analysis.

Seismic margins analyses typically calculate a high confidence/low probability of failure (HCLPF) for components or systems within the plant. This parameter represents an estimate of the seismic acceleration that the component could withstand and still perform its intended function. Past seismic margins analyses show that the HCLPF value for nuclear power plants is typically twice the design value. By determining the HCLPF for all the important components required to bring the plant to some stable state (typically hot shutdown for events initiated with the plant initially at power), the plant HCLPF, as well as the components or systems limiting the capability of a plant to survive a seismic event, can be determined. There are some limitations to a pure margins approach, some of which are magnified when there is a lack of design detail and no plant to physically examine. Limitations include the need to make assumptions about which systems and equipment would be required following a seismic event and the lack of accounting, in the methodology for non-seismic (i.e., random and human) failures. However, margins methods can be enhanced by using PRA insights to select success paths that likely would be used and to ensure that equipment reliability is considered in the analyses.

The staff intends to determine the adequacy (i.e., the robustness) of ALWR designs by using a modified margins analysis to identify seismic or other external event vulnerabilities. Insights about the seismic capability of the plant can be best obtained by merging PRA and margins approaches to take advantage of the strengths of each. This approach allows for a comprehensive and integrated treatment to understand the plant response to an earthquake. Plant logic models covering the various systems that could be used to prevent core damage are constructed typically by modifying internal events PRA models to include logic important to consideration of seismic failures. Neither site-specific nor generic seismic hazard curves are used as input for this approach. All significant operational sequences leading to safe shutdown (success paths) are identified using the event trees and fault trees based on fragility data for each component for each success path. The minimum HCLPF for each system determines the system and plant HCLPF. If the plant HCLPF is less than about twice the design ground motion zero period acceleration, a more detailed evaluation should be performed to determine if a vulnerability exists that should be strengthened. The HCLPF calculated in this manner provides a measure of the robustness of the plant in that it estimates with high confidence the earthquake ground motion the plant is expected to be able to survive without core damage. HCLPF calculations also provide insights about which components and systems limit the plant seismic capability.

A seismic PRA rather than the PRA-based margins approach described above was the primary approach used to evaluate the advanced boiling water reactor (ABWR) seismic design. The staff compared the hazard estimates derived by Lawrence Livermore National Laboratory (LLNL) using the historical earthquake method (NUREG/CR-4885, "Seismic Hazard Characterization of the Eastern United States: Comparative Evaluation of the EPRI and LLNL Studies") and by EPRI with the results obtained using the ABWR bounding seismic hazard curve. The ABWR bounding hazard curve was exceeded by the LLNL mean hazard curves for the Pilgrim, Seabrook, and Watts Bar sites. These three sites in the Eastern United States were selected because of their relatively high seismic hazard. The three LLNL curves were arithmetic mean hazard curves using input from five

ground motion experts. The staff used both the LLNL and EPRI seismic hazard estimates to quantify core damage frequency. The use of the LLNL hazard curves resulted in the prediction of much higher core damage frequencies than did the use of the EPRI hazard curve. However, the ABWR design was found to be capable of resisting earthquakes significantly larger than a safe shutdown earthquake of 0.3g based on an estimated plant HCLPF of about 0.6g using the seismic margins methodology.

Recent analyses of the seismic inputs to the LLNL hazard program suggest that uncertainties may be reduced in future LLNL results. As an example of reduced uncertainty in the LLNL estimates, the staff compared the LLNL hazard estimates of four (rather than five) ground motion experts. Results that included the estimates of the fifth ground motion expert were generally higher. Thus, using the estimates of only four ground motion experts might provide a means of showing the effect of reduced uncertainty. However, the LLNL results using the estimates of the four ground motion experts still exceeded the EPRI bounding curve. Thus, it is the staff's conclusion that the LLNL hazard results suggest the EPRI bounding curve for rock sites is not conservative.

In Section 3.3 of the DSER for Appendix A to Chapter 1, the staff identified quantitative uncertainty analysis for seismic events as an open issue. It also identified an open issue associated with the requirement in Section 3.3.2.4 to use a logarithmic standard deviation of 0.2 to reflect variability in the response spectrum. A seismic PRA is not required for seismic design certification. The staff will use the margins approach involving HCLPF values, as discussed above, in its review. Therefore, both of these DSER open issues are closed. However, when a specific site is selected for an evolutionary plant, the staff will review the combined license application to ensure that the plant-specific probabilistic seismic hazards analysis is in conformance with Section 2.5.2 of the SRP.

4 CONTAINMENT ANALYSIS

Section 4 of Appendix A to Chapter 1 of the Evolutionary Requirements Document discusses the containment analysis. The staff's conclusions on how the guidance should be modified are given here. However, it will still be necessary for the staff to review the containment analysis portion of a PRA submitted for the ALWR designs in some detail during its review of an individual application for FDA/DC.

4.1 Core-Damage-Sequence Binning

In Section 4.1 of Appendix A to Chapter 1 of the Evolutionary Requirements Document, EPRI states that core-damage sequences are expected to be binned (grouped). If core-damage bins are used, they must be defined so that all sequences within a particular bin lead to similar effects with respect to containment sequence and source term phenomena. EPRI requires that the definition of bins provide a means to ensure that the delineation of core-damage sequences is discriminated sufficiently to afford the proper level of coordination with the containment analysis. This is intended to provide a means of managing the number of accident sequences and to provide an additional means of gaining information needed for the in-plant analysis. The binning of sequences is an acceptable procedure to limit the number of containment analyses performed. It is necessary that, as EPRI states, all sequences within a bin lead to similar effects with respect to containment and source term phenomena. The staff will be evaluate sequence binning when it reviews an evolutionary design PRA as part of its review of an application for FDA/DC.

4.2 Containment System Analysis

Section 4.2 of Appendix A to Chapter 1 of the Evolutionary Requirements Document requires that a containment systems analysis be developed to explicitly account for any common-mode failures between the core-damage-prevention systems and the containment systems. To reduce the number of deterministic analysis runs necessary to develop the containment event tree branch point probabilities, EPRI made a simplifying assumption. The frequency dominant accident sequence for each plant damage state will be used to define in-plant phenomenological analysis parameters for use in determining containment performance source terms. EPRI cautions that the plant damage states must be sufficiently and uniquely defined to ensure that they adequately reflect the characteristics important to the containment response and release magnitudes in order to avoid introducing uncertainties that could otherwise be avoided. This procedure is a generally accepted part of PRA methodology. The staff will evaluate the actual groupings of the accident sequences into plant damage states when it reviews an evolutionary design PRA as part of its review of an application for FDA/DC.

4.3 Containment Isolation

Section 4.3 of Appendix A to Chapter 1 of the Evolutionary Requirements Document requires that containment penetrations be accounted for in the evaluation of containment leakage paths. EPRI requires that penetrations be analyzed for the following types of release pathways:

- failure to isolate normally open lines or normally closed lines that might be open at the time of an accident (e.g., because of the mispositioning of valves)

- leakage through a penetration where no pathways should be present (leakage past the seat of a closed valve, through an electrical penetration, past the seal of a personnel airlock, etc.)

EPRI allows containment penetrations to be screened from the analysis if they can meet one of the following criteria:

- small conditional probability of leakage or of failure to be isolated (i.e., less than 1E-3 per event)

- low consequence (e.g., release that must take place through a line that will remain filled with water throughout the accident)

- closed loop inside or outside the containment

- small in size (e.g., instrumentation lines)

EPRI's rationale is that not all containment penetrations have the potential to be important pathways for releases from the containment and the use of screening criteria is appropriate to focus the PRA effort on the penetrations that are most likely to be important. These simplifications are acceptable, but the staff will evaluate the actual implementation of these assumptions when it reviews an evolutionary design PRA as part of its review of an application for FDA/DC.

4.4 Containment Bypass

Section 4.4 of Appendix A to Chapter 1 of the Evolutionary Requirements Document requires that containment bypass sequences be assessed and include all connections to the reactor coolant system. EPRI states that containment bypass sequences can result in significant releases from the containment and have the potential to be important risk contributors. Bypass sequences that have been identified as important in past PRAs include

- steam generator tube rupture (PWR only)
- residual heat removal isolation failure
- high-pressure coolant injection (BWR only)
- core spray (BWR only)
- feedwater and main steam (BWR only)
- suppression system bypass (BWR only)

In the NUREG-1150 studies for Surry and Sequoyah, bypass sequences dominate early fatality risk. The staff agrees that containment bypass sequences are important and, therefore, agrees with the EPRI requirement to assess such sequences.

4.5 In-Plant Sequence Assessment

Section 4.5.2 of Appendix A to Chapter 1 of the Evolutionary Requirements Document states that the MAAP code will be the primary tool used to assess the thermal-hydraulic and other physical processes involved in the accident

progression. While recognizing the value of MAAP as an integrated code for severe-accident analysis, the staff concludes that this requirement places an undue reliance on the MAAP code and fails to establish a requirement for dealing with deficiencies of this code. More specifically, the staff is concerned that for best-estimate calculations, the MAAP code (or other codes for that matter) will not adequately treat certain physically important severe-accident phenomena, and that the models and assumptions made in the MAAP code will not be generally accepted by the experts in the field. In the DSER for Appendix A to Chapter 1, the staff stated that the Evolutionary Requirements Document should require that best-estimate MAAP calculations be supplemented by additional analyses that take into account alternative outcomes of potentially risk-significant issues and phenomena recognized by the nuclear safety community as containing large uncertainties. This was identified as an open issue in the DSER.

In response to the staff's DSER, EPRI proposed a modification to the Evolutionary Requirements Document to further clarify, by example, the situations in which use of codes other than MAAP might be more appropriate. These examples are "to develop realistic success criteria for core cooling, or to investigate specific phenomena that are not addressed by MAAP or that may be subject to large uncertainties." This modification partially addresses the staff's concern regarding use of the MAAP code for accident analysis. However, the document still fails to recognize the limitations and deficiencies of the MAAP code, such as those revealed through the recent industry/NRC MAAP 3.B familiarization effort.

In view of the remaining concerns regarding the use of codes for severe-accident analysis, the staff will review an application for FDA/DC against the codes most recently endorsed and approved by the NRC at that time, and will evaluate the acceptability of the plant designer's analyses on a case-by-case basis. This DSER open issue is closed.

The staff notes that the treatment of certain severe-accident phenomena within the scope of a PRA and the use of the MAAP code for this purpose is a related issue. This is discussed in Section 4.6 of Appendix A to Chapter 1 of the Evolutionary Requirements Document and in Sections 1.9 and 4.6 of this appendix.

4.6 Containment Event Analysis

Section 4.6.1 of Appendix A to Chapter 1 of the Evolutionary Requirements Document states that the containment event tree (CET) will comprise the important phenomenological issues associated with containment loading and/or source term evolution. As supporting rationale indicates, the CET provides an excellent means to identify and quantify important phenomena. The document lists issues and phenomena that should be considered by the ALWR designer in the development of the CET. In the DSER for Appendix A to Chapter 1, the staff stated that the CET should be expanded so that the adequacy of accident mitigation systems and of the estimation of risk measures related to public health and safety can be determined. This was identified as an open issue.

In a subsequent revision, EPRI added a notation that the issues to be considered are not limited to those listed. The staff concludes that the list of phenomena provides a reasonable characterization of the types of issues that should be addressed by a plant designer in the development of the CETs and

uncertainty analysis, but cautions that this set of phenomena may not necessarily be complete. For example, Section 6.6.2.6 of Chapter 5 of the Evolutionary Requirements Document states that a containment overpressure protection system (vent) system may be provided in ALWRs, yet no mention is made in Section 4.6.1 of Appendix A to Chapter 1 of the need to consider the impact of containment venting on containment success criteria or to treat containment venting in the CET. Therefore, the staff will evaluate how these issues were treated by the plant designer when it reviews an evolutionary design PRA as part of its review of an application for FDA/DC. This DSER open issue is closed.

Section 4.6.2 of Appendix A states that potentially important phenomena that are not currently addressed in the MAAP code will also be addressed in the containment event analysis and identifies five such phenomena in the rationale section. The staff agrees with this requirement, but noted in the DSER that the requirement should not apply only to phenomena that are not explicitly treated using the MAAP code. Rather, PRA analysis should consider all potentially risk-significant phenomena for possible inclusion in the containment event analysis. In response to this concern, EPRI modified the wording in Section 4.6.2 of Appendix A so as to not restrict consideration to only those phenomena that are not currently addressed in the MAAP code. The staff concludes that the revised guidance is acceptable, but notes that the phenomena identified in this section are not complete; for example, core debris coolability should also be addressed, since MAAP models for this phenomenon are highly questionable. The staff discusses this matter further in Section 1.9 of this appendix.

Section 4.6.3 of Appendix A states that the CETs will be quantified using best-estimate point values. The staff indicated in the DSER that the guidance in this section was insufficient, given that risk estimates may be dominated by uncertainties in modeling and assumptions related to severe-accident progression and phenomenology.

In response to the staff concerns in the DSER, EPRI proposed modifications to Section 4.6.3 of Appendix A that clarify, to some extent, the development of best-estimate values. The revised guidance states that for phenomenological events that could be important to the potential for releases and are subject to large uncertainty, the best estimates should reflect an appropriate assessment of expert opinion or other relevant input, in addition to the deterministic result obtained using MAAP. The document further states that the outcome of a deterministic code may not properly reflect the best estimate for some branch points, and that the deterministic outcome, together with other characterizations of the phenomena, should be used to arrive at a best estimate. An example provided in the document of one approach is "to develop a probability distribution for an event, reflecting the available inputs, with subjective weights assigned to each input. The mean value of such a distribution would then be used as the best estimate in the quantification process."

The staff considers that the proposed modifications partially address its concerns regarding Level 2 uncertainty analysis, but fall short in that they (1) fail to emphasize the importance of reflecting the range of possible issue outcomes in the containment analysis, particularly when an outcome may be bimodal in nature, such as debris coolability, and (2) fail to acknowledge

that it may be appropriate to perform a full, explicit treatment of uncertainties in selected Level 2 issues and phenomena. (This might be done as a limited extension of the Level 1 uncertainty analysis.) The staff discusses this matter in greater detail in Section 1.9 of this appendix.

Consideration of a range of possible uncertainties is especially important in the design phase to assess the adequacy of proposed severe-accident mitigation features and to minimize the potential impact of particular uncertainties through design modifications and/or accident management strategies. Such consideration is also important when the uncertainty range can envelop important measures of containment performance, such as the probability of containment failure given the (postulated) occurrence of certain poorly understood phenomena. The staff concludes that this uncertainty analysis must be performed in sufficient detail so as to provide reasonable assurance that potential design improvements and/or accident management strategies to deal with key uncertainties can be identified and their effectiveness estimated.

The staff's recent work discussed in NUREG-1150 represents the most extensive uncertainty analysis performed to date. However, the staff does not require a classical statistical analysis of this magnitude and believes that the safety-significant insights from a Level 2 analysis can be gained with approaches that require fewer resources than that of the NUREG-1150 approach. Toward this end, the staff supports use of NUREG-1150 data, modified where necessary, in vendor uncertainty analyses to reduce the number of parameters included in uncertainty analysis and/or to provide probability distributions, modified as necessary, for individual parameters. This was identified as an open issue in the staff's DSER for Appendix A to Chapter 1. Subsequent revisions of Appendix A by EPRI have not provided adequate guidance to vendors that addresses the staff's concerns. Therefore, the staff will review an individual application for FDA/DC as described above. This DSER open issue is closed.

4.7 Source Term Definition

Section 4.7 of Appendix A to Chapter 1 of the Evolutionary Requirements Document requires that the analyst use the most current version of the MAAP code available at the time for source term calculations. However, alternative codes may be used if justification is provided. In any case, EPRI states that an integrated model of the core melt and the containment is required to address generation, effects of steam inerting, containment geometry, and containment pressurization. The staff agrees with the use of integrated models, but has a number of technical concerns regarding the use of the MAAP code. (The staff's concerns regarding the use of the MAAP code are discussed in Section 4.5 of this appendix.) Accordingly, the staff will address source term definition during its review of an individual application for FDA/DC.

4.8 Plant Release Categories

Section 4.8 of Appendix A to Chapter 1 of the Evolutionary Requirements Document states that similar end points of the containment analysis may be grouped into release categories for use in the ex-plant consequence analysis. EPRI states that past PRAs have shown that CET end points may be grouped to simplify the analysis and reduce the number of ex-plant runs required. Elements to be considered in the grouping process include

- time of release
- duration of release
- energy of release
- types and amounts of isotope fractions released

The staff agrees that CET end points may be grouped. The type of grouping suggested by EPRI is similar to those used in past PRAs and is acceptable to the staff. The staff intends to review the details of the binning process as part of its review of an individual application for FDA/DC to confirm the process has been properly implemented.

Although the binning of CET end states is acceptable, the staff notes that Appendix A fails to provide guidance to ALWR designers regarding the use of the CET end state/release category information to develop risk measures related to containment performance, such as conditional containment failure probability and frequency of large releases. The staff will require ALWR designers to provide this type of information for their designs and will address this aspect of the analysis when it reviews an evolutionary ALWR PRA as part of its review of an application for FDA/DC. The staff's evaluation of EPRI's requirements regarding containment performance is provided in Chapter 5 of this report.

5 OFFSITE CONSEQUENCES

Section 5 of Appendix A to Chapter 1 of the Evolutionary Requirements Document discusses the assumptions for the offsite consequence analysis portion of the PRA.

5.1 Implementation of the Public-Safety Requirement

In Section 5.1 of Appendix A to Chapter 1 of the Evolutionary Requirements Document, EPRI requires that a complementary cumulative distribution function (CCDF) be developed for whole-body dose for a 1/2-mile radius. This must include all core-damage sequences with a frequency greater that 1E-8 per year from both internal and external initiators. EPRI proposes that the design will be considered to have met the EPRI risk goals if this CCDF falls outside the region bounded by a lower limit for frequency at 1E-6 per year and by a lower limit for the consequences of a 25-rem whole-body dose at 1/2 mile. The staff agrees that the CCDF is a well-accepted method of visually displaying risk curves. However, it displays only a point-estimate CCDF intended to approximate the mean CCDF and provides an incomplete picture of risk. In the DSER for Appendix A to Chapter 1, the staff stated that uncertainties must be propagated through the analysis, and a family of CCDFs must be displayed, each associated with their confidence limit, in addition to the mean CCDF. This was identified as an open issue.

The staff concludes that uncertainties must be propagated through the analysis, as discussed in Section 1.9 of this appendix, and a family of CCDF curves developed, each with their degree of belief (in addition to the mean CCDF curve). This information will be evaluated by the staff as part of the design certification review of each evolutionary ALWR PRA.

Although the Commission finds the goal of 1E-5 per year useful for designers and for potential customers of ALWR plant designs, it has stated that it does not intend to regulate advanced plants to a specific numerical core damage frequency goal. Although many measures of central tendency exist, the Commission has chosen "mean values" as the appropriate measure for safety goals and other Commission and staff risk targets. Therefore, when reporting core damage frequency and other risk measures, the staff concludes that "mean values" must be reported to the extent possible. Therefore, the staff will review an individual application for FDA/DC using the guidance stated above. This DSER open issue is closed.

5.2 Method for Offsite Consequence Analysis

In Section 5.2.1 of Appendix A to Chapter 1 of the Evolutionary Requirements Document, EPRI states that a reference site with the characteristics listed in Annex B will be used for calculating offsite consequences for the ALWR. Annex B contains meteorological data developed by EPRI to be representative of a bounding (80th percentile) site. This calculation of offsite consequences would be performed by each ALWR vendor to demonstrate that the plant design meets EPRI's goal for dose at the site boundary. The staff concludes that the use of a standardized set of meteorological site data is reasonable for the limited purpose of demonstrating that EPRI's design goal for dose at the site boundary has been met. The staff addresses the adequacy of the specific reference site data proposed by EPRI in Annex B of this appendix.

As part of FDA/DC for each evolutionary ALWR, the staff will require each ALWR vendor to provide an assessment of additional risk measures (such as person-rem and early and latent fatalities) to support the vendor's assessment of severe-accident mitigation alternatives for the ALWR design. Meteorological data alone are insufficient to calculate these additional risk measures and will need to be supplemented with bounding population data, such as those provided in Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Stations."

Section 5.2.2 of Appendix A to Chapter 1 states that either MACCS or CRAC2, or another suitable code, will be used for calculating offsite consequences. Although the CRAC2 code provides an acceptable characterization of the consequences of severe accidents, the MACCS code represents an improvement over CRAC2 and is preferred by the staff for cancer risk calculations. The present version of MACCS (Version 1.5) uses the results of the BEIR III study by the Committee on the Biological Effects of Ionizing Radiation, "The Effect on Population of Exposure to Low Levels of Ionizing Radiation," July 1980, in the calculation of health effects. However, the BEIR III study has been superseded by the BEIR V study. The BEIR V results indicate a higher cancer risk from low levels of ionizing radiation than did the earlier BEIR III report. The results of the BEIR V study must be taken into account in the calculation of health effects. The staff is preparing an addendum to NUREG/CR-4214, "Health Effects Models for Nuclear Power Plant Consequence Analysis," Revision 1, Part II, to address the modification of models resulting from recent reports on the health effects of ionizing radiation. The risk coefficients for fatal cancers would be approximately doubled or tripled by the model modifications. Until these modifications are incorporated into MACCS, the staff concludes that use of CRAC2 is acceptable, but that the effect of model differences must be taken into account in interpreting risk results. The staff will address this issue during its review of an individual application for FDA/DC.

6 UNCERTAINTY AND SENSITIVITY ANALYSES

In Section 6 of Appendix A to Chapter 1 of the Evolutionary Requirements Document, EPRI requires that sensitivity studies be performed for those issues or parameters that are judged to have relatively large associated uncertainty or that are particularly important to the PRA results. These sensitivity studies must include important aspects from each of the areas discussed below. EPRI indicates that the sensitivity studies may be qualitative or quantitative, depending on the nature of the issue being addressed, and are intended to provide adequate perspective with respect to uncertainty in the PRA results and the significance of potential contributors to risk.

In general, the staff accepts the scope defined for analysis (with a number of reservations stated below). However, it believes that the guidance in Section 6 is incomplete. For example, EPRI does not provide any details about the manner and range of variation for the parameters selected for back end sensitivity studies analysis. Moreover, the staff does not agree that sensitivity studies alone are in all cases an adequate substitute for uncertainty analysis. As discussed in Section 1.9 of this appendix, it is the staff's view that sensitivity analyses supplement uncertainty analysis by integrating areas of subjectivity in the analysis and by helping the analyst characterize major contributions to uncertainties in the results. The staff concludes that the uncertainty and sensitivity analyses required by Section 6 of Appendix A to Chapter 1 of the Evolutionary Requirements Document should be augmented as described below.

6.1 Analysis of Systems and Sequences

In Section 6.1 of Appendix A to Chapter 1 of the Evolutionary Requirements Document, EPRI lists areas that may be of particular importance with respect to the estimated frequency of core damage, including

- frequencies of rare initiating events that are important contributors to risk and any initiating events whose frequencies are assessed to be low relative to similar events for other nuclear power plants
- common-cause parameters, especially those whose assessment relied heavily on engineering judgment to compensate for a lack of experimental data
- human interactions following an initiating event
- seismic hazard

EPRI states that the list of areas represents those areas that may be most uncertain or are likely to be important to the frequency of core damage. In the DSER for Appendix A to Chapter 1, the staff stated that although it agreed that these areas should be considered, it believed there were other important areas of uncertainty that must explicitly be recognized. The staff concluded that the flood hazard (Section 3.2 of Appendix A to Chapter 1) must be included as well as failure rates for equipment items different from those used in current plants or subject to different operational conditions, such as advanced instrumentation and control components. This was identified as an open issue.

EPRI responded to the staff's DSER by indicating that the list was not intended to be exhaustive and by modifying the list to also include failure rates for equipment different from that used in current plants. The staff has considered the adequacy of EPRI's new guidance in terms of its consistency with (1) the scope of analyses that the staff will consider during its review of an application for FDA/DC, as described in Section 1.9 of this appendix; (2) the requirements placed on the PRA by the reliability assurance program; and (3) the guidance for external events provided in Section 3 of Appendix A to Chapter 1. As discussed in the following three sections, the staff concludes that EPRI's response is acceptable and will review an application for FDA/DC to ensure that these issues are adequately addressed.

Consistency of Scope

As discussed in Section 1.9 of this appendix, the staff will review an application for FDA/DC to ensure that ALWR vendor PRAs include a full uncertainty analysis for the Level 1 portion of the PRA, with uncertainties propagated from basic events, including initiating event frequencies, data, common cause/mode failure, success criteria, and human error. The staff concludes that the areas identified in Section 6.1 of Appendix A to Chapter 1 of the Evolutionary Requirements Document are consistent with the staff positions described in Section 1.9 of this appendix regarding treatment of uncertainties, provided the following areas are also addressed by the ALWR vendor in the uncertainty analysis:

- failure rates for equipment subject to operational conditions different from those in conventional PWRs and BWRs
- initiating events that are unique to evolutionary designs
- success criteria

The staff will evaluate the adequacy of the vendor's treatment of uncertainty in each of the aforementioned areas when it reviews a design-specific PRA as part of its review of an application for FDA/DC.

Importance Analyses for Input to Reliability Assurance Program

The staff requires the ALWR vendors, in addition to systematically assessing the significance of uncertainties in risk results, to use the ALWR PRAs to provide insights into equipment that should be included in the design-specific reliability assurance program (RAP) and for which each vendor will formulate guidance. Such equipment will be identified using importance analysis of the PRA model.

EPRI has provided guidance in Section 6 of Chapter 1 of the Evolutionary Requirements Document on conducting a RAP; however, this guidance lacks specificity regarding how a utility is to determine what specific equipment should be included in the RAP. The staff will review applications for FDA/DC to ensure that ALWR vendors provide a list of equipment they believe should be included in the RAP, and requires that the ALWR vendors provide guidance to combined license applicants on how the applicants should determine what additional equipment should be included in the program, on the basis of the as-built plant. Although the specific NRC review criteria for the RAP are still evolving, it is clear that the design-specific PRA should be used to

provide a technical basis for many aspects of the RAP and should be integrated into RAP development and application. Accordingly, the staff requires that ALWR vendors calculate and submit (for their RAP) the results of importance analyses for each of the basic events in the PRA, using "importance measures" to rank systems, structures, and components (SSCs). Appropriate "importance measures" should be chosen so that they, as a group, provide all the needed information on the SSCs' effect in controlling and reducing risk. The risk-based ranking of SSCs for the RAP can be achieved by using importance measures such the risk achievement (or risk increase worth), the risk reduction worth, or equivalent measures of importance such as Fussel-Vesely and Birnbaum. The risk achievement (or risk increase worth) importance measure is of special interest in RAP because it measures the relative importance of features that contribute to achieving the risk level assumed in the PRA and toward which reliability assurance activities should be directed. The risk reduction worth is of special interest in ranking activities aimed at reducing risk, such as modifications in plant operations or design.

The staff concludes that vendors must provide two sets of analyses as part of an application for FDA/DC - the first to reflect the relative importance of systems, components, and actions to core damage from internally initiated events, and the second to reflect the importance when all initiators are considered (i.e., internal and external events combined). The use of two sets is to prevent the possibility that insights from considering external initiators may swamp insights from considering internal initiators, since the staff believes that external initiators have larger uncertainties than many internal initiators and since the ALWR designs have significantly reduced the absolute value estimates of frequency of core damage from internal initiators. Importance analyses for external events analyzed with bounding or margins-type analyses will have to be evaluated in a qualitative manner.

The risk significance of equipment important to the Level 2 analysis may not be readily identified through traditional importance analyses, which focus only on core damage frequency. Accordingly, in screening systems, structures, and components for inclusion in the RAP, the vendor (and subsequently the combined license applicant) must also consider equipment important to accident progression, containment performance, and releases. The staff would expect the vendor to develop a systematic method that uses the Level 2 portion of the PRA, and the associated sensitivity and uncertainty analyses, to perform this additional assessment.

In general, the ALWR vendor must (1) provide insight into what systems, structures, and components should be included in the RAP and (2) give guidance to combined license applicants on how to determine which systems, structures, and components should be included in the RAP on the basis of the as-built plant. Vendor guidance needs to address situations in which bounding or non-probabilistic analyses are used by the ALWR vendor, since importance analyses may not accurately reflect the importance of equipment. Similarly, there may be areas and equipment that are in the plant or are site specific and that are safety significant, but are not modeled in the PRA (e.g., structures, passive components, a heater for a borated tank, or heat tracing for piping with highly borated water). Importance analysis would provide no insights in these areas. The staff will evaluate the adequacy of the vendor's importance analyses at the FDA/DC and combined license stages for each evolutionary design PRA.

External Events

The staff intends to use high confidence/low probability of failure (HCLPF) values (both for the plant and by event sequence) obtained from the seismic margins analysis to help make regulatory and licensing decisions. Additional regulatory and licensing insights will be gained from evaluating sequences ranked by HCLPF values. Therefore, the staff will not require that hazard curve uncertainties be addressed in the design-specific PRA or its updates, as discussed in Section 3.3 of this appendix.

A severe-accident fire analysis must include at least a qualitative evaluation of uncertainties including fire initiation frequency and the chance that smoke, hot gases, and/or fire suppressant will cross 3-hour fire barriers that separate divisions.

A severe-accident internal flooding analysis must include at least a qualitative evaluation of the uncertainties, including flood initiation frequency, flow rates, detection failure and detection timing, and migration of water to other divisions or floors.

For external flooding, as discussed by the staff in Section 3.2 of this appendix, a detailed uncertainty analysis is not required because the return period uncertainties dwarf those of random equipment failure or equipment capacities. For external flooding, if the expected frequency with which river floods will exceed the probable maximum flood (PMF) is 1E-5 per year or greater, the combined license applicant should perform an analysis of the conditional probability of being unable to maintain core cooling, containment integrity, and reactivity control. It is expected that there would be adequate warning time in case of a flood so that the plant would be shut down before the flood exceeded the PMF.

6.2 Assessment of Containment Response

In Section 6.2 of Appendix A to Chapter 1 of the Evolutionary Requirements Document, EPRI lists the following important parameters that could affect the frequency of serious releases that must be considered in the sensitivity studies:

- parameters related to hydrogen burn
- core debris coolability
- pressure capacity of the containment and the location and size of pressure-induced failure of the containment
- parameters that could affect high-pressure melt ejection
- parameters associated with the production of combustible gas outside the reactor vessel
- operator actions that could affect accident progression

EPRI states that these are among the most important uncertainties that could affect the assessment of conditional frequency of releases due to core-damage accidents. In the DSER for Appendix A to Chapter 1, the staff agreed that

uncertainties in these and other parameters were important but stated that sensitivity studies alone were not adequate, and must be supplemented by a full, quantitative uncertainty analysis.

The staff has reconsidered the issue of uncertainty analyses and concludes that a full, quantitative uncertainty analysis is not required for the Level 2 portion of the PRA analysis, which covers the aforementioned areas. Instead, the staff requires (1) the implementation of a systematic process for identifying issues and phenomena of greatest risk significance for the advanced designs and (2) a more thorough treatment of those issues and phenomena and their associated uncertainties as part of the Level 2 analysis, for example, in the containment event trees. (See Section 1.9 of this appendix for additional discussion.)

The staff concludes that the list of issues and phenomena in Section 6.2 of Appendix A is a reasonable set, but notes that this set is not necessarily complete. For example, containment bypass and aerosol plugging may be important for some ALWR designs, but these are not included on the list.

The staff will address the acceptability of the ALWR designer's treatment of severe-accident issues and phenomena as part of the design certification review for each ALWR design. The adequacy of the analysis will be judged on the basis of the completeness of issues considered, reasonableness of parameter ranges and screening criteria for sensitivity analyses, and the methodology for decomposing and propagating the range of potential issues outcomes in the analysis. The analysis must provide the staff reasonable assurance that the PRA reflects the significance of key actions, events, and phenomena for the plant design, as well as the effectiveness of the accident mitigation features. This DSER open issue closed.

6.3 Analysis of Source Terms

In Section 6.3 of Appendix A to Chapter 1 of the Evolutionary Requirements Document, EPRI lists the following parameters that could affect the magnitude of accidental radionuclide releases that must be investigated in appropriate sensitivity studies:

- the effectiveness of containment scrubbing mechanisms
- the form of iodine and the corresponding amount of iodine available for release

EPRI states that these are some of the most important parameters associated with the behavior of fission products in the reactor coolant system and in the containment that could affect the amount released to the atmosphere after containment failure. The staff indicated in the DSER that it was appropriate to consider these parameters in sensitivity studies, but stated that sensitivity studies alone were insufficient and must be supplemented by a quantitative uncertainty analysis.

The staff has reconsidered the issue of uncertainty analyses and concludes that a full, quantitative analysis of uncertainty is not required for the Levels 2 and 3 portions of the PRA.

As stated by the staff in Section 1.9 of this appendix, ALWR vendors must use a more systematic process for identifying risk-significant issues of relevance to the ALWR plant designs. This would include the conduct of sensitivity analysis to screen issues and identify those with significant impact on risk. Although the thrust of this systematic process is on Level 2 issues, it is also important that plant designers identify and assess those Level 3 issues that may result in a shift of accident frequencies from relatively begin accident release classes to the category of "large release," and consider the need to incorporate uncertainties in these issues in the context of the full uncertainty analysis.

In a February 7, 1991, response to an NRC question concerning plant certification issues contained in SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," EPRI stated that the plant designers must confirm as part of the PRA that the source terms for representative accident sequences for their actual standard plant designs are bounded by the physically based source term in the Evolutionary Requirements Document. However, no mention of the need for PRA analysts to perform this source term assessment is made in Appendix A to Chapter 1. This was identified as an open issue in the DSER.

Responding to this DSER open issue in a letter dated May 1, 1992, EPRI stated that it has modified Appendix A to Chapter 1 of the Evolutionary Requirements Document and has left the choice of using either the current source term specified in Atomic Energy Commission document TID 14844 or the updated physically based source term to the designers. At this time, both General Electric and ASEA Brown Bovlai/Combustion Engineering have indicated that their evolutionary designs can meet 10 CFR Part 100 dose limits with the TID 14844 source term. However, the vendors have also indicated that they may wish to apply the revised source term to their designs when it becomes available. Should a designer elect to use the physically based source term for licensing calculations, EPRI stated that it will develop and incorporate a requirement in the Evolutionary Requirements Document for the designer to compare the physically based source term with and the PRA source term at that time. The staff concludes that this is acceptable. Therefore, this DSER open issue is closed.

7 HUMAN RELIABILITY ANALYSIS

Guidance on the analysis of human interactions is provided in Section 2.9 of Appendix A to Chapter 1 of the Evolutionary Requirements Document. This section specifies that the EPRI Systematic Human Action Reliability Procedure (SHARP) analysis framework will be used for considering human interactions in the PRA and requires that the analysis deal explicitly with the following:

(1) definition of human actions
(2) screening for importance
(3) task breakdown
(4) representation in relation to systems logic models
(5) iteration between human and hardware modeling
(6) quantification
(7) documentation

EPRI's SHARP analysis requires that human interactions be placed into the following categories:

- interactions before an initiating event (Type A)
- actions related to the initiating events (Type B)
- interactions following an initiating event, including actions dictated by procedures (Type CP), and actions to recover equipment or systems to terminate a sequence (Type CR)

High-level guidance on the required scope of the human reliability analysis (HRA) and the review, classification, and quantification of each of these types of human interactions is provided in subsequent sections of Appendix A.

The staff, in its DSER for Appendix A to Chapter 1, identified several major shortcomings in the EPRI guidance. These included inadequacies in the document with regard to the following:

- incorporating HRA considerations throughout the PRA requirements process and providing mechanisms to ensure that credible and auditable results are obtained
- providing guidance and approaches for the conduct of function, task, and timeline analyses
- providing systematic guidelines for selecting and for evaluating person-centered, task-centered, and environment-centered performance shaping factors (PSFs)
- providing information on state-of-knowledge PSF evaluation tools to ensure that the PRA results are credible, auditable, and consistent across sequences and systems
- providing information on state-of-knowledge quantification methods

In a letter dated January 24, 1992, regarding these concerns, EPRI proposed modifications to the Evolutionary Requirements Document. The staff has reviewed Appendix A and EPRI's responses to the DSER in terms of the (1) scope and objectives of the required analysis, (2) framework or process for validating the HRA to reflect the detailed design, and (3) guidance and requirements for conducting the specific elements of the HRA. The results of this evaluation are provided below.

7.1 Scope and Objectives of HRA

In its DSER for Appendix A to Chapter 1, the staff identified inadequacies in incorporating HRA considerations throughout the PRA requirements process and providing mechanisms to ensure that credible and auditable results are obtained as an open issue. EPRI's proposed revisions to Section 2.9 of Appendix A are primarily aimed at further clarifying the intended scope and objectives of the HRA. The thrust of the modifications is to more clearly reflect that the plant design will not be complete when the HRA is performed by the vendor and that because of the state of the plant design, there will be limitations on the analysis beyond those normally encountered in conducting an HRA. EPRI states that when the HRA is performed there will be no operating or emergency procedures for the plant, no control room, and no directly applicable experience with a like plant. As a result, EPRI's stated objectives of the HRA are "to obtain a nominal assessment of the human interactions to support the overall risk assessment, and to ensure that the assessment is accomplished in a consistent and traceable fashion." The revisions establish that the assessment of human interactions will be sufficient to accomplish the following:

- identify the types of human interactions that may be important to risk for the ALWR design
- provide a nominal quantification of the human interactions sufficient to support the overall assessment of core damage frequency and frequency of severe release
- provide a mechanism to investigate the potential effects of varying the reliability assessed for the human interactions
- establish a framework for performing a more extensive HRA when the design progresses to the state of an actual plant, with operating procedures, layout, etc.

Consistent with the reduced scope, EPRI has redefined those items that the analysis must explicitly address to consist of the following in lieu of Items (1) through (7) in Section 7 above:

- plant logic model construction
- quantification
- analysis of recovery actions
- internal review

The staff recognizes that it is not possible to conduct a meaningful detailed HRA in the absence of such specific items as emergency operating procedures, control room design layout and staffing, and plant simulators. Accordingly, the scope, objectives, and level of detail of an analysis performed at this

stage of plant design would be distinctly different from, and more limited than, those associated with an HRA for a completed plant design. The staff considers the HRA objectives set forth by EPRI in the revised Evolutionary Requirements Document to be reasonable for a "scoping analysis" or "approximate quantification" for a partially completed plant design. However, the staff will review applications for FDA/DC or a combined license to ensure that vendors and applicants provide a more detailed and defensible treatment of human reliability after the design details have been developed and before plant operation. Such details include the design of the control room and man-machine interface, control room staffing, emergency procedures, and operator training. This process is discussed in greater detail in Sections 7.2 and 7.3 below. This DSER open issue is closed.

7.2 Process for Validating the HRA

Section 2.9 of Appendix A to Chapter 1 of the Evolutionary Requirements Document provides no specific requirements or guidance concerning refinement and validation of the scoping HRA to reflect the details of the final design, such as the use of advanced technology and automation, and its impact on human performance and the HRA.

The staff will review applications for FDA/DC to ensure that the vendor for each ALWR design establishes the process and criteria by which the combined license applicant will confirm the adequacy of the HRA treatment and, if appropriate, upgrade the analysis. It is the staff's expectation that this HRA implementation plan will establish an iterative process (involving a limited number of iterations) by which information developed as the control room design matures (and possibly other aspects of the design) is assessed against the HRA assumptions, and conversely, HRA insights are factored into the control room design. Inputs to the HRA would come from several elements of the plant design process, such as the control room function analysis, task analysis, and man-in-the-loop testing. These inputs are expected to be qualitative in nature, but would provide a basis for judging certain quantitative aspects of the HRA. Outputs from the HRA would provide insights into other elements, such as the development of the man-machine interface and plant procedures. The staff will evaluate the HRA implementation plan or process when it reviews an ALWR design-specific PRA as part of its review of an individual application for FDA/DC.

7.3 Guidance on Specific Elements of the HRA

Use of New Technologies

Increased automation (automation of tasks traditionally performed by an operator) and enhanced decision aids in ALWRs will result in a shift of the operator's role from a direct manual controller to a supervisory controller and system monitor. The shift in role away from direct control is typically viewed as positive from a reliability standpoint, since the operator is considered one of the more unpredictable components in the system. It has been observed in other industries, however, that although some errors may be reduced or eliminated, such a change has frequently been associated with a shift of human error to higher levels in the system that are more difficult to detect and quantify. Examples of such errors include (1) errors in attempting to recover from the failure of an automated system; (2) errors in the setup of automated systems, such as keying the wrong information or data into an

automated system; (3) actions taken in response to a false alarm (provided by an intelligent system and not adequately checked); and (4) failure to properly monitor the automated system (loss of vigilance because of overreliance on the automated system). The potential for these "new" types of errors that can occur in an advanced system must be considered in the human action modeling and HRA for ALWRs.

The staff has reviewed the degree to which the Evolutionary Requirements Document addresses, or at least recognizes, the potential for advanced technologies (e.g., digital displays, touch screens, automatic decision aids) anticipated in ALWR plant designs, and their implications for the design-specific PRA. These include the impact on human performance due to increased automation, changes in the human-system interface and methods of operator interaction with the system, and advanced control room data management systems. The staff concludes that EPRI's guidance is deficient insofar as taking note of and providing contingencies for dealing with new technologies anticipated and currently being incorporated into ALWR designs. However, it is unclear if state-of-knowledge HRA modeling, data gathering, and performance quantification tools are adequate to handle anticipated designs and the technologies they embrace, such as automated decision aids. This suggests the need for more empirical approaches to modeling and quantification, such as part-task prototyping and full-scale control room prototypes.

In the staff's view, the impact of advanced technologies on the HRA is an issue that is best addressed by the vendor and combined license applicant during the iterative process of designing the control room, rather than as part of the scoping HRA. In this regard, the staff requires that combined license applicants pay particular attention to the effects of system automation, decision aids, and artificial intelligence on human performance and errors as part of control room design process. A possible method for ascertaining the impact of the advanced technologies would be to perform accident simulations or walk-through, talk-through exercises with plant operators as part of the development of the control room prototype and of the detailed control room design. A final evaluation of the impact of advanced technology on the HRA must be made after the control room design and plant-specific simulator are available, as part of the subsequent effort to validate the HRA. In either case, the focus of these evaluations would be on observing the impact of the advanced features on operator performance and error potential, and confirming that potentially significant errors are included in the HRA.

The staff will review applications for FDA/DC to ensure that ALWR vendors define the process and acceptance criteria to be used by the combined license applicant to address the impact of new technologies on human performance. This process should ensure that potentially significant new human errors (i.e., not considered in the scoping HRA) are identified during the detailed development of the control room and addressed early in the design process either by adoption of alternative design approaches or by modification of the HRA to better reflect the potential for additional errors. The process should also provide guidance to the combined license applicant regarding the use of the plant-specific simulator in validating the assumptions and models in the HRA for the PRA. The staff will evaluate this process when it reviews an ALWR design-specific PRA as part of its review of an application for FDA/DC.

Function, Task, and Timeline Analyses

The staff has evaluated the degree to which Appendix A to Chapter 1 establishes requirements and provides guidance on the conduct of function, task, and timeline analyses. These analyses provide a mechanism for characterizing human actions and human action sequences and identifying potential problems in operator performance. This is especially important in areas where individual cognitive errors and combinations of human errors are possible during dynamic accident sequences. In its DSER for Appendix A to Chapter 1, the staff identified the lack of guidance in this area as a deficiency in Appendix A to Chapter 1. In response, by letter dated January 24, 1992, EPRI indicated that function, task, timeline, and link analyses were inappropriate, given the state of the plant design at the design certification stage.

The staff concludes that Appendix A of Chapter 1 still does not provide adequate guidance in this area. Moreover, the document fails to establish a process for performing a more detailed assessment of human performance using appropriate analysis techniques once the plant design details have been established. The staff considers the EPRI guidance on the modeling of human interactions to be adequate for scoping-type analyses of human performance. However, the staff will review applications for a combined license to ensure that a reassessment of the HRA is performed before plant operation that is based on function and task analyses that reflect the final design. (These analyses are typically performed as part of the detailed control room design process and the development of operating procedures.) The staff will review an application for FDA/DC to ensure that the ALWR vendor has developed guidance for a process and criteria to be used by the combined license applicant to perform this reassessment. The staff will evaluate the process and criteria when it reviews an ALWR design-specific PRA as part of its review of an application for FDA/DC. The DSER open issue related to function, task, timeline, and link analyses is closed.

Generic Data Sources

The staff has reviewed the degree to which generic data sources are identified in Appendix A to Chapter 1. These data are needed as bounding and anchor values for quantifying individual human actions and more complex human action sequences involving several interdependent actions by individuals or groups of individuals. In the DSER for Appendix A to Chapter 1, the staff stated that EPRI had committed in a letter dated November 7, 1990, to add other sources of generic data in addition to the THERP (Technique for Human Error Rate Prediction) handbook (NUREG/CR-1278, "Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Plant Applications"). This was identified as a confirmatory issue in the DSER.

The staff has confirmed that Appendix A has been revised to reference a number of additional sources of generic data, including the following:

- more recent work by Swain performed as part of the NRC Accident Sequence Evaluation Program (NUREG/CR-4772, "Accident Sequence Evaluation Program Human Reliability Analysis Procedure")

- the revised SHARP document (EPRI NP-7183-SL)

- more recent work by Parry (EPRI TR-100259) performed as part of the EPRI Operator Reliability Experiment project

The staff considers the data sources identified in Appendix A to Chapter 1 to be an acceptable but minimal set. It expects ALWR vendors to support the use of the data contained therein by comparisons with data from additional sources (e.g., other peer-reviewed HRAs). The NRC-sponsored Nuclear Computerized Library for Assessing Reactor Reliability provides one such source of both human and hardware component failure data that might be used by ALWR plant designers to support their PRA and HRA. The staff will assess this aspect of the HRA when it reviews an ALWR design-specific PRA as part of its review of an application for FDA/DC, in conjunction with the issues of quantification and performance shaping factors. This confirmatory issue is closed.

Performance Shaping Factors and Evaluation Tools

The staff has evaluated the degree to which Appendix A to Chapter 1 provides (1) systematic guidelines for selecting person-centered, task-centered, and environment-centered performance shaping factors (PSFs) to ensure that the task actions are adequately characterized and (2) information on PSF evaluation tools to ensure that the HRA results are credible, auditable, and consistent across sequences and systems. In the DSER for Appendix A to Chapter 1, the lack of guidance in this area was identified as an open issue.

In response to staff concerns, EPRI modified Section 2.9 of Appendix A to Chapter 1 to refer to more recent work by Parry (EPRI TR-100259) for quantification of human interactions. The recent work provides two distinct methods of quantification, either of which can be used depending on the level of information available. One method uses a time-reliability correlation, and the other focuses on assessing PSFs such as the nature and content of procedures and level of training. These methods provide approximations of human performance derived from data that implicitly reflect the effects of PSFs without attempting to characterize individual contributing factors. EPRI also contended in its response that detailed treatment of PSFs would require information that could not be specified until after an actual plant was built, or at least a very good simulator was available.

The staff concludes that the guidance and approach in Appendix A to Chapter 1 of the Evolutionary Requirements Document regarding PSFs are adequate performing scoping analyses of human interactions, but insufficient with regard to the types of analyses that are needed after ALWR design details have been established and the plant-specific simulator is available. Explicitly identifying and evaluating specific PSFs is a key factor in any HRA. Without a clear understanding of the PSFs as they pertain to a particular plant, the credibility of the quantitative results is reduced, as is the potential to gain insights into remedial actions required for reducing human contributions to risk. Accordingly, the staff will review applications for FDA/DC to ensure that combined license applicants will perform a further assessment of PSFs and their impact on the HRA results after the ALWR control room design details have been established. The staff will also review applications for FDA/DC to ensure that ALWR vendors have developed guidance on the process and criteria to be used by the combined license applicants to assess the impact of PSFs on HRA results. Information on preparing this type of guidance is available from documentation developed by EPRI (Reports NP-309, "Human Factors Review of Nuclear Power Plant Control Room Design"; NP-2411, "Human Engineering Guide

for Enhancing Nuclear Control Room"; and NP-1982, "Evaluation of Proposed Control Room Improvements Through Analysis of Critical Operator Actions") and that developed under a variety of control room design, training, and operating procedures programs of the NRC. The staff will evaluate this process when it reviews an ALWR design-specific PRA as part of its review of an application for FDA/DC. The DSER open issues concerning PSFs and their evaluation tools are closed.

Quantification Methods

The staff has reviewed the degree to which Appendix A to Chapter 1 provides guidance on state-of-knowledge quantification methods to ensure that the HRA results are credible, auditable, and consistent across sequences and systems.

In the DSER for Appendix A to Chapter 1, the staff indicated that the guidance in Appendix A did not adequately address the numerous quantification methods that have been developed and tested over the past 4 to 6 years by various domestic and international organizations involved in HRA. As mentioned in the two preceding subsections, EPRI has revised Appendix A to refer to more recent work performed by EPRI (Parry et al.). The more recent work by Swain (NUREG-/CR-4772) has also been referenced. Methods in either of these references could be used for assessing the nominal probabilities of human interactions for the ALWR PRAs and should produce results that can be readily traced. In either case, it would be necessary to document the timeline for the human interactions, as well as any other relevant features.

The staff concludes that the guidance in Appendix A still provides a less than complete picture of alternative quantification methods, some of which address dynamic modeling and cognitive error issues more comprehensively than the methods identified in the Evolutionary Requirements Document. However, the staff also recognizes that these alternative techniques are not a panacea for the difficulties in human performance quantification, because large uncertainty and variability are inherent in the application of any of these techniques. This has been demonstrated in EURATOM's Human Reliability Benchmarking study.

Most critical to the quality of an HRA is that all significant human actions are represented in the analysis and that the relative importance of human actions in the risk profile of the plant is determined. The particular numerical values assigned to human actions, in and of themselves, are of second order importance. With regard to the adequacy of the HRA quantification, ALWR vendors are required to further assess human error probability values as part of the sensitivity and uncertainty analyses called for in Sections 1.4 and 6.1 of Appendix A to Chapter 1. The staff will review applications for a combined license to ensure that applicants reassess the adequacy of the HRA quantification as part of the HRA validation process after the control room design details have been established. In this context, the staff would expect combined license applicants to make use of the control room prototype and plant-specific simulator to the extent necessary to confirm the adequacy of the HRA in terms of completeness of human interactions represented in the analysis and reasonableness of assumed human error probability values. With the design-specific considerations described above, the staff concludes that the EPRI guidance on quantification is adequate for the purpose of ALWR HRAs and this DSER open issue is closed.

Documentation and Reporting

The staff has reviewed the degree to which Appendix A to Chapter 1 provides guidelines for developing and maintaining "running" documentation on HRA work in progress and for reporting the results of the HRA.

The SHARP framework discussed in Section 2.9 of Appendix A provides an outline for documenting the HRA. Although general, the outline does provide a mechanism for documenting the HRA to allow users and reviewers to judge the degree to which each of the following has been achieved:

- credible quantitative results (point estimates, uncertainty bounds, confidence intervals)
- quantitative and qualitative results that are fully auditable, that is, provide traceable relationships between the error probability estimates and human factors information on which the estimates are based
- qualitative results that can support causal analyses of problem areas and attendant remedial actions
- quantitative and qualitative data that are reproducible and lend themselves to issues that transcend individual sequences and plant systems
- state-of-knowledge computational methods, data, and procedures for conducting HRA within the context of a PRA
- contingencies for dealing with new technologies anticipated and currently being incorporated into ALWR designs

The staff concludes that the guidelines provided by EPRI for developing and maintaining documentation on HRA work in progress and for reporting the results of the PRA are acceptable.

8 CONCLUSION

The staff concludes that the EPRI requirements established in Appendix A to Chapter 1 of the Evolutionary Requirements Document for conducting a PRA 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 design-specific PRAs will be adequate. Individual applicants for FDA/DC referencing the Evolutionary Requirements Document will be required to submit a design-specific PRA for review by the staff.

Therefore, the staff concludes that Appendix A to Chapter 1 of the Evolutionary Requirements Document specifies requirements that, subject to resolution of the identified vendor- and utility-specific items, if properly translated into a design and constructed and operated in accordance with the NRC regulations in force at the time the design is submitted, should result in a facility 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. 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.

ANNEX A - RELIABILITY DATA BASE FOR ALWR PRAs

Annex A of Appendix A to Chapter 1 of the Evolutionary Requirements Document describes the development of the initiating event frequencies and component reliability data to be used in developing the PRA. Section A1 of Annex A outlines the methods used in obtaining initiating event frequencies for loss-of-coolant accidents and for most transient events. Section A2 of Annex A describes the treatment of the frequency and recovery of losses of offsite power in greater detail.

Section A3 of Annex A summarizes the sources of data used to arrive at the recommended hardware failure rates, maintenance unavailabilities, and common-cause factors.

Annex A, Part 2, contains the results of a survey of component failure rates for evolutionary ALWRs.

Section A2, "Loss of Offsite Power"

This section provides background information on how the loss-of-offsite-power frequencies were established for the reliability data base to be used in the ALWR PRAs. In the DSER for Appendix A to Chapter 1, the staff stated that the total loss-of-offsite-power frequency established by EPRI for the ALWR (0.0077 loss per site-year) is an order of magnitude less than the long-term historical average for total loss of offsite power at U.S. nuclear plants (0.07 loss per site-year) and is approximately 23 percent of the 3-year average (0.033 loss per site-year for 1986, 1987, and 1988) found in Nuclear Safety Analysis Center Report NSAC/147, "Losses of Off-site Power at U.S. Nuclear Power Plants: Through 1989." In the DSER, the staff recommended that EPRI work with the staff in resolving differences between NSAC/147 and the staff's data for loss-of-offsite-power events and identified this as an open issue.

In a letter dated January 26, 1992, EPRI stated that data related to loss of offsite power will be revised to include consideration of the NRC Office for Analysis and Evaluation of Operational Data (AEOD) data after the policy issue on alternative source of power for non-safety loads is resolved with the staff. The staff concludes that EPRI's response is acceptable and will review individual applications for FDA/DC to ensure that design-specific PRAs use data that accurately characterize the nature of the offsite power system, including a loss-of-offsite-power frequency of 0.033 per site-year. Therefore, this DSER open issue is closed.

Annex A, Part 2, "Survey of Component Failure Rates for Evolutionary ALWRs"

Item No. 65, Transformer (high voltage): fails to continue operating

Under this item, failure rates of high-voltage transformers taken from several sources are provided. The failure rate selected for ALWR PRAs is 1.2E-6 failure per hour. In the DSER for Appendix A to Chapter 1, the staff noted that the failure rate appeared to be low for main step-up transformers judging

from the number of reports the staff has received on main transformer failures. In response, EPRI modified Annex A to address main step-up transformers explicitly with a failure-to-continue-operating frequency of 5.4E-6 per hour. This value is acceptable to the staff. Therefore, this issue is closed.

ANNEX B - ALWR REFERENCE SITE

Annex B of Appendix A to Chapter 1 of the Evolutionary Requirements Document originally included reference site data on meteorology, population density and distribution, and evacuation and sheltering. When the DSER for Appendix A to Chapter 1 was issued, the staff understood that EPRI was evaluating whether to modify or delete the reference site data. In the DSER, the staff stated that it would review any revisions upon submittal, as appropriate, and identified this as an open issue. Consistent with the desire to decouple the plant design from siting considerations, all but the meteorological data were deleted in a subsequent revision. The staff concludes that this is acceptable because the population and evacuation data are not needed to estimate dose at the site boundary. The meteorological data were also updated to be more representative of a bounding (80th percentile) site.

By letter dated April 9, 1992, EPRI submitted a summary of the methods and assumptions used in the development of the ALWR 80th percentile meteorological data base. The staff has reviewed this information and concludes that the methods and assumptions used by EPRI in the development of the ALWR 80th percentile meteorological data base are consistent with current practice and methodology. Therefore, this DSER open issue is closed. As part of the FDA/DC review for each ALWR PRA, the staff will evaluate the site data used by the vendor in estimating dose at the site boundary to ensure that these data are representative of most potential ALWR sites.

ANNEX C - COMPONENT SEISMIC CAPACITIES

Annex C of Appendix A to Chapter 1 of the Evolutionary Requirements Document describes the process by which generic seismic capacities were estimated for the evolutionary ALWR designs. These generic capacities, defined in terms of median spectral accelerations, were generated by reviewing available seismic PRAs, seismic fragility analyses, seismic margins analyses, and generic equipment ruggedness data.

EPRI states that the component capacities do not represent the highest or lowest capacities that can be identified in the literature. Rather, the values represent what EPRI considers reasonable capacities for the ALWR design. If the fragility values generated from the use of these data result in overly conservative results, a component-specific capacity may be developed that is based on design-specific or vendor-supplied information.

Annex C also describes the methodology and assumptions used in developing generic component fragilities.

The generic fragilities presented in this document may not be applicable to some of the advanced instrumentation and control components that may be used in ALWR reactors. The staff cautions ALWR vendors to this effect and requires that they justify the use of generic fragilities in their seismic analyses.

CHAPTER 1, APPENDIX B, "LICENSING AND REGULATORY REQUIREMENTS AND GUIDANCE"

1 INTRODUCTION

This appendix of the SER documents the NRC staff's review of Appendix B, "Licensing and Regulatory Requirements and Guidance," to Chapter 1 the Evolutionary Requirements Document through Revision 3. Appendix B to Chapter 1 was prepared, under the project direction of EPRI and the ALWR Utility Steering Committee, by Grove Engineering, Inc.; MPR Associates, Inc.; S. Levy Inc.; TENERA, L.P.; and EPRI.

Although Appendix B to Chapter 1 was not submitted as part of the original Evolutionary Requirements Document, it was included in Revision 1, which EPRI submitted on September 7, 1990. This appendix consolidates the resolutions proposed by EPRI for its proposed optimization subjects and certain unresolved and generic safety issues. Revisions 2, 3, and 4 of the Evolutionary Requirements Document were docketed on April 26 and November 15, 1991, and April 17, 1992, respectively. EPRI submitted additional information regarding this appendix by letters dated February 12 and December 6, 1991, and March 3, 1992.

Because EPRI had not originally submitted Appendix B to Chapter 1, the staff did not develop a DSER for it. However, many of the topics now contained in Appendix B were discussed throughout the 13 chapters of the original document. Accordingly, the staff's original review of many of these topics was discussed in the corresponding chapters of the DSER.

1.1 Review Criteria

Section 1 of Volume 1 of this report describes the approach and review criteria used by the staff during its review of Appendix B to Chapter 1 of the Evolutionary Requirements Document.

1.2 Scope and Structure of Appendix B to Chapter 1

Appendix B to Chapter 1 of the Evolutionary Requirements Document is a compilation of the regulatory requirements and guidance that EPRI believes are applicable to the design of evolutionary ALWRs. In addition, it gives EPRI's justification for deviations from current criteria and its proposed resolution of unresolved and generic safety issues that are considered applicable to the evolutionary ALWR design.

The key topics addressed in the Appendix B review include EPRI-proposed requirements for

- compliance with the Commission's regulations and regulatory guidance
- EPRI-proposed optimization subjects
- resolution of certain unresolved and generic safety issues
- resolution of policy issues

1.3 Regulatory Requirements and Guidance

In Section 1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document lists all NRC regulations and guidance that EPRI believes are currently applicable to LWR designs. EPRI states that this section identifies the regulatory requirements (Table B.1-1) and guidelines (Table B.1-2) it believes are applicable to the design of the ALWR at the level of detail consistent with that of the Evolutionary Requirements Document when the document is completed. Tables B.1-3 and B.1-4 identify those requirements and guidelines that EPRI believes are not applicable to the ALWR design.

Section 10.2 of Chapter 1 of the Evolutionary Requirements Document states that the ALWR will comply with the NRC regulatory requirements and guidance in effect on January 1, 1990, consistent with the commitments in Section 1 of Appendix B to Chapter 1. EPRI states that these requirements and guidance include applicable Commission regulations in Title 10 of the Code of Federal Regulations (10 CFR), general design criteria, NRC policy statements, regulatory guides, NUREG-0800 ("Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants") and other documentation that resolves unresolved and generic safety issues. Although the staff understands EPRI's need to "freeze" the requirements its addresses to those in effect on January 1, 1990, as stated in Section 10 of Chapter 1 of this report, the staff expects that the design certification applications will be in compliance with the Commission's regulations and guidance that are applicable and in effect at the time the certification is issued. In addition, issue resolutions that are different from those arrived at during the staff's review of the Evolutionary Requirements Document may be developed as the staff completes its reviews of the detailed design information provided in the applications for final design approval and design certification (FDA/DC), and as these designs are litigated in the design certification hearings. Therefore, the staff concludes that the ALWR plant designers will comply with the issue resolutions adopted by the NRC staff during its reviews of applications for FDA/DC in accordance with the requirements of 10 CFR Part 52. The staff will evaluate this compliance during its review of an individual application for FDA/DC.

Section 10.2.4 of Chapter 1 states that the plant designer will provide an ALWR design that is consistent with the disposition of the regulatory requirements and guidance identified in Section 1 of Appendix B. EPRI defines the commitment specified in Appendix B as follows:

- Comply

  The "comply" designation indicates that the ALWR design will fully comply with all regulatory requirements and guidance provided by the reference.

- Optimization Subject

  "Optimization subjects" are proposals initiated by EPRI to deviate from regulatory requirements. EPRI proposes to resolve these issues by providing technically supportable alternatives to current regulatory requirements. EPRI specifies that the ALWR design will comply with all regulatory requirements and guidance for a regulatory item associated with an optimization subject, except as described in Section 2 of Appendix B to Chapter 1 of the Evolutionary Requirements Document.

The staff concludes that, except as discussed throughout this report, Table B.1-1 through Table B.1-4 of Appendix B are generally acceptable for the evolutionary plant designs. The staff's review of the entire Evolutionary Requirements Document provides a clearer picture of the extent to which the document complies with the Commission's regulations.

1.4 Outstanding Issues

As stated previously, Appendix B to Chapter 1 was not submitted as part of the original Evolutionary Requirements Document package. Therefore, the staff did not develop a DSER for this appendix and no outstanding issues were associated with it. However, many of the topics now contained in Appendix B were discussed throughout the 13 chapters of the original document. Accordingly, the staff's original review of many of these topics was discussed in the corresponding chapters of the DSER.

On September 7, 1990, EPRI submitted Revision 1 of the Evolutionary Requirements Document, which included Appendix B to Chapter 1. During its review of Appendix B to Chapter 1, the staff identified several items that fall into the category of an outstanding policy issue on which the staff has taken a position, but for which the Commission has not had the opportunity to provide guidance.

The outstanding issues, with references to appropriate sections of this appendix given in parentheses, are listed below. The designators in front of each issue provide a unique identifier for each issue. The letter "E" indicates that the issue applies to the evolutionary plant design. The first number designates the chapter in which it is identified, and the letter that follows designates the appendix. The letter "O" designates that it is an open issue. The final number is the sequential number assigned to it.

Open Issues

E.1B.O-1 impact of the elimination of the operating-basis earthquake from the design process (2.1.1, Item IV.A of Annex A, and Item I.M of Annex C, Item C of Annex D)

E.1B.O-2 applicability of industry codes and standards (2.1.1 and Item II.A of Annex C)

E.1B.O-3 tornado design basis (2.1.2 and Item II.F of Annex C)

E.1B.O-4 main steamline classification (2.3.1.1 and Item II.E of Annex C)

E.1B.O-5 simplification of postaccident sampling system (2.3.2 and Item II.I of Annex C)

E.1B.O-6 containment leak rate testing (2.5.1 and Item II.H of Annex C)

E.1B.O-7 source term (2.5.2.1, 2.5.2.2, Item I.B of Annex A, and Item I.A of Annex C)

E.1B.O-8 seismic hazard curves (Item II.C of Annex C)

E.1B.O-9 leak before break (Item II.D of Annex C)

E.1B.O-10 containment bypass (Item II.G of Annex C)

E.1B.O-11 prototyping (Item II.K of Annex C)

E.1B.O-12 reliability assurance program (Item II.M of Annex C)

E.1B.O-13 defense against common-mode failures in digital instrumentation and control systems (Item A of Annex D)

E.1B.O-14 analysis of external events beyond the design basis (Item B of Annex D)

E.1B.O-15 control room annunciator reliability (Item G of Annex D)

Confirmatory Issues

None

1.5 Vendor- or Utility-Specific Items

The vendor- or utility-specific items, with references to appropriate sections of this appendix given in parentheses, are listed below. The designators in front of each issue provide a unique identifier for each issue. The letter "E" indicates that the issue applies to the evolutionary plant designs. The first number designates the chapter in which it is identified, and the letter that follows designates the appendix. The letter "V" designates that it is a vendor- or utility-specific item. The final number is the sequential number assigned to it.

E.1B.V-1 compliance of FDA/DC applications with Commission's regulations and guidance (1.3)

E.1B.V-2 issue resolution for FDA/DC reviews (1.3)

E.1B.V-3 elimination of missile provisions (2.1.2)

E.1B.V-4 dynamic seismic analysis of main steam piping and condenser (2.3.1.1 and Item II.E of Annex C)

E.1B.V-5 main steamline classification (2.3.1.1 and Item II.E of Annex C)

E.1B.V-6 seismic analysis and plant walkdown of turbine building (2.3.1.1 and Item II.E of Annex C)

E.1B.V-7 plateout considerations for main steam piping and valves (2.3.1.2 and Item III.F of Annex C)

E.1B.V-8 reactor pressure vessel level instrumentation system (2.4.1)

E.1B.V-9 source term (2.5.2.1, 2.5.2.2, Item I.B of Annex A, and Item I.A of Annex C)

E.1B.V-10 compliance with Branch Technical Position MTEB 6.1 (2.5.2.2)

E.1B.V-11 fission product cleanup analysis (2.5.2.2)

E.1B.V-12 deletion of charcoal adsorbers (2.5.2.2)

E.1B.V-13 dedicated containment vent penetration (2.5.3 and Item I.K of Annex C)

E.1B.V-14 decoupling of operating-basis earthquake (OBE) from safe shutdown earthquake (SSE) in seismic design of structures (Generic Safety Issue A-40) (3.2.7)

E.1B.V-15 deletion of OBE damping values in seismic design of structures (Generic Safety Issue A-40) (3.2.7)

E.1B.V-16 use of algebraic sum method for modal combination of high-frequency modes for vibratory loads (Generic Safety Issue A-40) (3.2.7)

E.1B.V-17 use of spectral peak shifting techniques in lieu of spectral broadening (Generic Safety Issue A-40) (3.2.7)

E.1B.V-18 plant-specific design and arrangement of control systems (Generic Safety Issue A-47) (3.2.9)

E.1B.V-19 conformance to 10 CFR 50.34(f) hydrogen control requirements (Generic Safety Issues A-48 and 121) (3.2.10 and 3.2.46)

E.1B.V-20 reliability of emergency diesel generators (Generic Safety Issue B-56) (3.2.14)

E.1B.V-21 resolution of Generic Safety Issues 2 and 110 (3.2.18 and 3.2.42)

E.1B.V-22 resolution of Generic Safety Issue 15 (3.2.19)

E.1B.V-23 independent reactor coolant pump seal cooling during station blackout (Generic Safety Issue 23) (3.2.20)

E.1B.V-24 resolution of Generic Safety Issue 24 (3.2.21)

E.1B.V-25 design details on threaded fasteners (Generic Safety Issue 29) (3.2.22)

E.1B.V-26 reduction of biofouling in open-cycle service water and component cooling water systems (Generic Safety Issue 51) (3.2.23)

E.1B.V-27 resolution of Generic Safety Issue 57 (3.2.24)

E.1B.V-28 resolution of Generic Safety Issue 73 (3.2.26)

E.1B.V-29 equipment classification and vendor interface for reactor trip system components (Generic Safety Issue 75) (3.2.27)

E.1B.V-30 2-week requirement for corrective maintenance (Generic Safety Issue 75) (3.2.27)

E.1B.V-31 preventive maintenance and surveillance program for reactor trip breakers (Generic Safety Issue 75) (3.2.27)

E.1B.V-32 resolution of Generic Safety Issue 76 (3.2.28)

E.1B.V-33 cooldown rate in natural convection cooldown analysis (Generic Safety Issue 79) (3.2.29)

E.1B.V-34 low-density storage racks in spent fuel pool for most recently discharged fuel (Generic Safety Issue 82) (3.2.30)

E.1B.V-35 plant-specific design and arrangement for control room heating, ventilating, and air conditioning (HVAC) system (Generic Safety Issue 83) (3.2.31)

E.1B.V-36 design of emergency filter units (Generic Safety Issue 83) (3.2.31)

E.1B.V-37 design details for control room capacity following a design-basis accident (Generic Safety Issue 83) (3.2.31)

E.1B.V-38 design details for control room HVAC systems in the smoke removal mode (Generic Safety Issue 83) (3.2.31)

E.1B.V-39 resolution of Generic Safety Issue 87 (3.2.33)

E.1B.V-40 adequacy of low-temperature overpressure protection design (Generic Safety Issue 94) (3.2.34)

E.1B.V-41 adequacy of BWR water level redundancy (Generic Safety Issue 101) (3.2.37)

E.1B.V-42 interfacing system design details (Generic Safety Issue 105) (3.2.39)

E.1B.V-43 inservice testing programs and technical specifications for appropriate pressure isolation valves (Generic Safety Issue 105) (3.2.39)

E.1B.V-44 resolution of Generic Safety Issue 106 (3.2.40)

E.1B.V-45 environmental qualification and inservice inspection and testing of large-bore hydraulic snubbers (Generic Safety Issue 113) (3.2.43)

E.1B.V-46 use of prestressed concrete containments (Generic Safety Issue 118) (3.2.44)

E.1B.V-47 reliability, operability, and on-line testability of protection system final actuation contacts (Generic Safety Issue 120) (3.2.45)

E.1B.V-48 operator training program and emergency operating procedures related to initiating feed-and-bleed cooling (Generic Safety Issue 122.2) (3.2.50)

E.1B.V-49 auxiliary feedwater analyses (Generic Safety Issue 124) (3.2.52)

E.1B.V-50 operational aspects of electrical power reliability (Generic Safety Issue 128) (3.2.56)

E.1B.V-51 resolution of Generic Safety Issue 130 (3.2.57)

E.1B.V-52 resolution of Generic Safety Issue 132 (3.2.58)

E.1B.V-53 resolution of Generic Safety Issue 135 (3.2.59)

E.1B.V-54 resolution of Generic Safety Issue 142 (3.2.60)

E.1B.V-55 resolution of Generic Safety Issue 143 (3.2.61)

E.1B.V-56 resolution of Generic Safety Issue 151 (3.2.62)

E.1B.V-57 assessment of safety service water system failure modes and contributions to core damage frequency and identification of dominant accident sequences (Generic Safety Issue 153 (3.2.63)

E.1B.V-58 resolution of Generic Safety Issue HF 4.4 (3.2.64)

E.1B.V-59 resolution of Generic Safety Issue HF 5.1 (3.2.65)

E.1B.V-60 resolution of Generic Safety Issue HF 5.2 (3.2.66)

2 PLANT OPTIMIZATION SUBJECTS

Plant optimization subjects are proposals initiated by EPRI to deviate from regulatory requirements. EPRI proposes to resolve these issues by providing technically supportable alternatives to current regulatory requirements. Table 1.2 of Volume 1 of this report lists EPRI's proposed plant optimization subjects and their applicability to the evolutionary and passive plant designs. These issues are identified for the evolutionary plant design in Section 2 of Appendix B to Chapter 1 of the Evolutionary Requirements Document. The staff's evaluation is given in this section under the corresponding sections in Appendix B to Chapter 1 of the Evolutionary Requirements Document. The regulatory departure analyses discussed in Section 4 and provided in Annexes A through C of this appendix provide additional information on these subjects.

2.1 Issues Related to Chapter 1 of the Evolutionary Requirements Document

2.1.1 Operating-Basis Earthquake and Dynamic Analysis Methods

In Section 2.3.C of Revision 0 of Chapter 1 of the Evolutionary Requirements Document, EPRI proposed to define the peak ground acceleration of the operating-basis earthquake (OBE) to be one-third that of the safe shutdown earthquake (SSE). In Section 2.3.C of the DSER for Chapter 1, the staff stated that it had not completed its review of this matter. As a result of discussions with the staff and the Advisory Committee on Reactor Safeguards, EPRI modified its position. In Section 2.1.1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document, EPRI now proposes to eliminate the OBE as a design-basis event and to use alternative criteria for dynamic analysis methods. In SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," dated January 12, 1990, and the draft Commission paper, "Issues Pertaining to Evolutionary and Passive Light Water Reactor and Their Relationship to Current Regulatory Requirements," issued on February 27, 1992, the staff stated that it agreed that the OBE should not control the design of safety-related systems. As a result, the staff is in the process of rulemaking for Appendix A to 10 CFR Part 100 to decouple the OBE from the SSE in siting considerations. The staff is also evaluating the possibility of redefining the OBE in order to satisfy the OBE's function without explicitly analyzing responses. This change would diminish the role of the OBE in design by establishing a level that, if exceeded, would require that the plant be shut down for inspection activities. The staff agrees in principle with EPRI regarding the deletion of the OBE from plant design.

The implementation of this optimization subject will have a broad impact on many technical issues and will result in changes to many existing staff positions. For example, Section 2.1.1 of Appendix B to Chapter 1 suggests changes to Appendix A to 10 CFR Part 100, 13 regulatory guides, and 12 sections of the Standard Review Plan (SRP, NUREG-0800). Most of the technical issues pertaining to this optimization subject have been addressed by the staff as a part of its review of Section 4 of Chapter 1 of the Evolutionary Requirements Document. As discussed in various parts of Section 4 of Chapter 1 of this report, the staff is not in complete agreement with the Evolutionary Requirements Document on several of the dynamic analysis issues. In addition, in Section 4.4.3 of Chapter 1 of this report, the staff states that

it is still evaluating the impact of the elimination of the OBE from the design process. Therefore, because of the broad nature of this plant optimization subject and the lack of complete agreement between the staff and EPRI on many of the issues involved, the staff concludes that the information in Section 2.1.1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document is not completely acceptable. Although the staff has identified its position on this matter in the draft Commission paper issued on February 27, 1992, its approach to resolving this issue has not been reviewed by the Commission and, therefore, does not represent an agency position. Therefore, the staff regards this is an open issue that must be satisfactorily resolved before it can complete its review of the Evolutionary Requirements Document. The staff's review of applicable issues is contained in Section 4 of Chapter 1 of this report. This optimization subject is discussed further in this report in Section 4.4.3 of Chapter 1 and under Item IV.A of SECY-90-016 contained in Annex A, Item I.M of the draft Commission paper issued on February 27, 1992, contained in Annex C, and Item C of the draft Commission paper issued on July 6, 1992, contained in Annex D of this appendix.

The concerns about the use of the OBE in the design that were raised by the staff in its letter dated April 24, 1991, are no longer applicable to its review of the Evolutionary Requirements Document because of its position regarding this optimization subject. However, as discussed in the DSER dated September 1987 for Chapter 1, certain issues related to the treatment of earthquake cycles for piping and equipment fatigue evaluations, seismic anchor motion effects, and concrete structure design also need to be resolved.

EPRI also proposes to reduce the conservatism in current seismic analysis techniques such as the damping values, algebraic combination of high-frequency modes, broadening of the response spectrum, spectral shifting analysis technique, multiple response spectrum analysis technique, and classification of piping stresses. Most of these issues are discussed in this report (see Chapter 1).

In its letter dated April 24, 1991, the staff questioned the appropriateness of classifying seismic stress as secondary stress. EPRI agreed that seismic stress was not self-limiting and was a load-controlled stress. In Revision 3 of the Evolutionary Requirements Document, EPRI deleted the sentence classifying seismic stress as secondary stress. The staff concludes that this modification is acceptable and this issue is closed.

Revision 3 of Section 2.1.1 of Appendix B to Chapter 1 indicates that ASME Code Cases N-451 and N-462 provide alternative rules on allowable seismic stresses for Class 1 and Class 2/3 piping. It also states that these code cases have not been accepted by the staff and that the ASME Code Committee is implementing changes to the code. However, until these code cases or the changes in the ASME Code are accepted by the staff, current rules acceptable to the staff should be followed. The staff's position on the applicability of industry codes and standards for the ALWR designs is provided under Item II.A of the draft Commission paper issued on February 27, 1992, contained in Annex C of this appendix. Basically, the issue of allowable seismic stresses for systems and components is not associated with civil structures. Although the staff has identified its position on this matter in the draft Commission papers of February 27 and July 6, 1992, its approach to resolving this issue has not been reviewed by the Commission and, therefore, does not represent an

agency position. This is an open issue that must be satisfactorily resolved before the staff can complete its review of the Evolutionary Requirements Document.

2.1.2 Tornado Design

Section 2.1.2 of Appendix B to Chapter 1 and Section 4.4.3.3.8 of Chapter 1 of the Evolutionary Requirements Document state that the plant designer will use American National Standards Institute/American Nuclear Society (ANSI/ANS) 2.3, "Standard for Estimating Tornado and Extreme Wind Characteristics at Nuclear Power Sites," to define tornado effects, using a probability of 10E-6 per year. This is an exception to Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants." The staff's evaluation of this issue is given in this report in Section 4.5.2 of Chapter 1 and under Item II.F of the draft Commission paper contained in Annex C of appendix. Although the staff has identified its position on this matter in the draft Commission paper of February 27, 1992, its approach to resolving this issue has not been reviewed by the Commission and, therefore, does not represent an agency position. This is an open issue that must be satisfactorily resolved before the staff can complete its review of the Evolutionary Requirements Document.

Elimination of Missile Provisions

Section 2.1.2.3 of Appendix B to Chapter 1 of the Evolutionary Requirements Document states that certain missile provisions, such as labyrinths in heating, ventilating, and air conditioning inlets and exhausts will be eliminated. EPRI states this elimination will lower construction costs and enhance maintenance.

Regulatory Guide (RG) 1.117, "Tornado Design Classification," and SRP Section 3.5.2, "Structures, Systems, and Components To Be Protected From Externally Generated Missiles," describe a method acceptable to the staff for identifying those structures, systems, and components that should be protected from the effects of the design-basis tornado, including tornado missiles. Depending on the nature and source of the externally generated missiles, protection may be provided by missile barriers for individual components, by placing independent redundant systems in compartments located in a missile-protected structure, or by a subgrade location at a sufficient depth. The staff concludes that the EPRI-proposed elimination of certain missile provisions may be acceptable, provided sufficient justification is provided on a case-by-case basis regarding compliance with RG 1.117 and SRP Section 3.5.2. The staff concludes that EPRI's generic justification of reduction in cost and maintenance is not sufficient for eliminating missile-protection measures. The staff will evaluate this matter during its review of an individual application for FDA/DC.

2.2 Issues Related to Chapter 2 of the Evolutionary Requirements Document

There were no plant optimization subjects associated with Chapter 2 of the Evolutionary Requirements Document.

2.3 Issues Related to Chapter 3 of the Evolutionary Requirements Document

2.3.1 BWR Main Steamline Isolation Valves and Leakage Control and Classification of Main Steamline of Boiling-Water Reactor

In Section 2.3.1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document, EPRI proposes to eliminate the BWR main steam isolation valve leakage control system and provide an alternative leakage pathway (i.e., the main steamline and the condenser) to the main condenser downstream of the isolation valves in the event of a loss-of-coolant accident (LOCA). This issue is also related to Section 3.3.2 in Appendix B to Chapter 1, Section 3.4.1.5 in Chapter 2, Sections 5.3.3 and 5.4.1.5 in Chapter 3, Section 1.2.3 in Chapter 5, and Section 3 in Chapter 13 of the Evolutionary Requirements Document. In its letters dated May 17 and August 29, 1991, the staff stated its positions on the quality group and seismic classifications of the BWR main/extraction steam system, main steamline, main turbine system, condenser, and related non-seismic structures that resulted from the proposed elimination of the main steam isolation valve leakage control system. The objective of these staff positions is to establish adequate measures to ensure that the above systems and structures will maintain their structural integrity during and following an SSE. Although this issue involves information in Chapters 1, 2, 3, 5, 6, and 13, for the sake of continuity, the basis for the staff's positions and the staff's evaluation of all of EPRI's responses are discussed below.

Main Steamline Classification

Background

The main steamlines in BWR plants contain dual quick-closing main steam isolation valves (MSIVs). These valves function to isolate the reactor system in the event of a break in a steamline outside the primary containment, a design-basis LOCA, or other events requiring containment isolation. Although the MSIVs are designed to provide a leaktight barrier, it is recognized that some leakage through the valves will occur. The current technical specification limit for MSIV leakage is typically 11.5 standard cubic feet per hour (scfh) per valve. Operating experience indicates that degradation has occasionally occurred in the leaktightness of MSIVs and the specified low leakage has not always been maintained.

Because of recurring problems with excessive leakage of MSIVs, the staff recommended in RG 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants," the installation of a supplemental leakage control system (LCS) to ensure that the isolation function of the MSIVs is in accord with the specified limits.

In response to the MSIV leakage concerns, the BWR Owners Group (BWROG) commissioned a program of studies to determine the causes of high leakage rates and the means to eliminate them. The results of these studies were submitted to the NRC in General Electric proprietary reports NEDO-31643P (dated November 1988) and NEDO-31858P (dated February 1991), both entitled "Increasing Main Steam Isolation Valve Leakage Rate Limits and Elimination of Leakage Control Systems."

In addition, the NRC has recently reviewed the desirability of the MSIV LCS requirement. Because the LCS may not be effective for controlling MSIV leakage rates greatly in excess of technical specification limits because of limitations in its design, the staff established Generic Issue C-8, "Main Steam Line Valve Leakage Control Systems," to address these concerns for currently operating plants.

The staff's position on the EPRI-proposed resolution for both evolutionary and passive BWR designs follows. It is based on the assumed capability of the main steam piping (including its associated piping to the condenser) and the condenser to remain structurally intact so they can act as a holdup volume for fission products.

Evaluation

The MSIVs do not provide a leaktight containment pressure boundary as intended in the plant design. Although substantial progress has been made in recent years to identify the causes of the leakage and to reduce the amount of leakage, the current typical technical specification limit of 11.5 scfh per valve is still difficult to achieve when the valve is rapidly closed against full-flow conditions at full reactor pressure and temperature.

The current assumption for operating plants in calculating the dose reference values specified in 10 CFR Part 100 is based on a conservative assumption that the leakage allowed by the technical specifications of 11.5 scfh per valve is released directly into the environment. No credit is currently taken for the pressure integrity of the main steam piping and condenser.

The proposed approach developed by EPRI and the BWROG would allow higher leakage limits and would take credit for the main steam piping and condenser to plate out the fission products following core damage. In this way, the main steam piping (and its associated piping) and the condenser would be used to mitigate the radiological consequences of an accident that could result in potential offsite exposures comparable to the dose reference values in 10 CFR Part 100.

Seismic Qualification: Appendix A to 10 CFR Part 100 requires that structures, systems, and components necessary to ensure the capability to mitigate the radiological consequences of accidents that could result in exposures comparable to the dose guideline exposures of Part 100 be designed to remain functional during and after a safe shutdown earthquake (SSE). Thus, the main steamline, portions of its associated piping, and the main condenser are required to remain functional if the SSE occurs. As such, these components are required to be analyzed to demonstrate their capability to maintain their structural integrity. Appendix A to 10 CFR Part 100 requires that the engineering method used to ensure that the safety functions are maintained during and after an SSE involve the use of either a suitable dynamic analysis or a suitable qualification test.

Non-Nuclear Experience: In Appendix D to NEDO-31858P, BWROG submitted a report to the staff entitled "Performance of Condensers and Main Steam Piping in Past Earthquakes," Report No. 50032.02-R-01 dated September 1990 by EQE Engineering. The report provided a summary of data on the performance of non-seismically designed main steam piping and condensers in past earthquakes

around the world. These data were used to evaluate the capability of main steam piping and condensers in typical General Electric (GE) Mark I, II, and III BWRs in the United States to withstand design-basis earthquakes.

The EQE report showed that piping and condensers exhibited substantial seismic ruggedness even when they were not designed for earthquake loadings. No failures of main steam piping were found. The report concluded that (1) the possibility of significant failure of GE BWR main steam piping or condensers in the event of a design-basis earthquake in the Eastern United States was highly unlikely and (2) any such failure would be contrary to the large body of historical earthquake experience data.

As a result of its review of the EQE report for condensers, the staff concludes that, in general, the experience data base is applicable to BWR condensers in U.S. plants. Overall, condensers in nuclear plants are similar to condensers in non-nuclear plants. However, because the pipe wall thickness is typically about 1 inch for main steam piping in BWRs, whereas the pipe wall thickness is typically much greater than 1 inch, and as much as 4 inches, for main steam piping in fossil plants, the staff concludes that the generic study for the main steam piping is not appropriate for BWRs.

Piping Failure Modes: The staff recognizes that welded piping systems exhibit seismic ruggedness even under loadings several times greater than a design-basis earthquake loading. The staff has also investigated the actual failure modes of piping due to earthquake loadings. In NUREG-1061, Volume 2 Addendum, "Summary and Evaluation of Historical Strong-Motion Earthquake Seismic Response and Damage to Above-Ground Industrial Piping," dated April 1985, the staff found that the observed failure of piping can be divided into three primary categories:

(1) failure of pipe due to excessive displacement of attached equipment

(2) failure of branch piping due to excessive displacement of attached piping mains

(3) failure of piping associated with loss of pipe supports

Another observed category of pipe failure (beyond the scope of NUREG-1061) is that caused by the failure of the supporting or enclosing building which, in turn, causes the failure of the pipe.

Classification of Main Steamline of BWRs: Because of the lack of evidence provided by the EQE report to support generic acceptability of the BWR main steam piping to withstand an SSE, the staff concludes that seismic analyses are required to ensure that the failure modes described above will not occur in BWR main steam piping. The staff also concludes that no undue burden exists for the ALWRs to require its main steam piping and branch lines to the first isolation valve to be classified as safety-related, seismic Category I piping with the appropriate quality assurance requirements imposed by Appendix B to 10 CFR Part 50 before the plant is constructed. Appendix A to SRP Section 3.2.2, "System Quality Group Classification," provides guidelines on the safety classification of BWR main steam piping. The SRP recommends that the main steamline from the second isolation valve up to, but not including,

the turbine stop valve including branch lines to the first valve be classified as Quality Group B (Safety Class 2). RG 1.29, "Seismic Design Classification," designates such piping as seismic Category I.

The staff's position is that the main steam piping from the outermost isolation valve up to the seismic interface restraint and branch lines up to the first closed valve must conform to Appendix A to SRP Section 3.2.2 and RG 1.29. The main steamline from the seismic interface restraint up to, but not including, the turbine stop valve (including branch lines to the first normally closed valve) must be classified as Quality Group B and inspected in accordance with applicable portions of ASME Code, Section XI, "Inservice Inspection," but may be classified as non-seismic Category I if it has been analyzed using a dynamic seismic analysis to demonstrate its structural integrity under SSE loading conditions. However, all pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 are applicable to ensure that the quality of the piping material is commensurate with its importance to safety during both operational and accident conditions.

The seismic interface restraint must provide a structural barrier between the seismic Category I portion of the main steamline in the reactor building and the non-seismic Category I portions of the main steamline in the turbine building. The seismic interface restraint will be located inside the seismic Category I building. The main steamline in the turbine building must be classified as non-seismic Category I to be consistent with the classification of the turbine building. Therefore, the quality and safety requirements proposed by the staff for the main steamline from the outermost isolation valve up to the turbine stop valve are equivalent to the staff guidelines in Appendix A to SRP Section 3.2.2 and RG 1.29.

Classification of Piping Associated With the Main Steamline: The BWROG has identified several interconnected alternative leakage pathways to allow MSIV leakage to drain to the condenser. Of these, the BWROG has proposed to use the drain line downstream of the MSIVs to provide a non-safety-grade leakage pathway to the condenser. In accordance with the BWROG recommendations, Section 2.3.1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document proposes the use of the main steamline drains for MSIV leakage control.

However, GE has indicated that the bypass piping of the main steamline is another appropriate leakage pathway that the designer may use for MSIV leakage control. The staff concludes that use of either pathway is an acceptable approach to resolving this issue, provided the chosen pathway is seismically analyzed to withstand SSE loading conditions, as discussed below. In addition, because these pathways are interconnected, the designer will have to seismically qualify all pathways up to the first normally closed valve to ensure this piping maintains its integrity during an SSE.

For ensuring the integrity of the MSIV leakage pathway (i.e., drain line or bypass line) from the first valve to the main condenser hotwell, the staff and EPRI both agree that preventing structural failure of the piping and hotwell would provide assurance that leakage from the MSIVs following a design-basis accident would not result in potential offsite exposures that exceed the dose guidelines of 10 CFR Part 100. The staff's position regarding the classification of the MSIV leakage pathway between the first normally closed valve and the condenser hotwell and the hotwell itself is as follows.

The staff proposes that (1) the MSIV leakage pathway from the first valve up to the condenser inlet and (2) the piping between the turbine stop valve and the turbine inlet not be classified as safety related or as seismic Category I, but should be analyzed using a dynamic seismic analysis to demonstrate their structural integrity under SSE loading conditions.

EPRI's proposal to use the seismic experience data base in lieu of a seismic analysis for the line from the first valve of the MSIV leakage pathway to the condenser is not consistent with the staff's position discussed above and is, therefore, not acceptable. Although this portion of the MSIV leakage pathway does not have to be seismic Category I, the experience data base does not provide adequate assurance of seismic integrity for such piping. In addition, EPRI's proposal to use the simplified seismic analysis procedure described in RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," is not acceptable for the first valve of the MSIV leakage pathway up to and including the condenser. To satisfy Item II.k of SRP Section 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment," this piping and the condenser should be analyzed to the SSE loading condition using the same criteria as those that are applicable to seismic Category I items. The guidelines in RG 1.143 allow a simplified analysis using only the OBE loading conditions and are unacceptable. Therefore, the staff will evaluate this piping and the condenser during its review of an individual application for FDA/DC in accordance with the positions in Appendix A to SRP Section 3.2.2 and Regulatory Guide 1.29.

Turbine Stop Valve and Associated Piping and Structures: The staff's evaluations of EPRI's responses relative to the turbine stop valve, the piping between the stop valve and the turbine, and the structures associated with this issue are given in Section 3.1.1 of Chapter 13 of this report.

Condenser Design: Because the Evolutionary Requirements Document does not address the condenser design details and no ALWR plant has been constructed, the staff cannot evaluate the applicability of the generic seismic experience data base for the ALWR condenser. Thus, it requires that the condenser be seismically analyzed by the combined license applicant to ensure that it is capable of maintaining its structural integrity during and after an SSE. The condenser is not required to be classified as seismic Category I or as a safety-related component. The staff concludes that the additional safety-related and seismic Category I requirements (e.g., the quality assurance requirements in Appendix B to 10 CFR Part 50 or the inservice inspection requirements in the ASME Code) are excessive and unnecessary for ensuring the structural integrity of the condenser under seismic loadings with no significant increase in safety if they were to be imposed. However, applicants seeking to take advantage of the main steam piping, its associated piping, and the condenser as an alternative leakage path will be required to submit a request for an exemption from the quality assurance requirements of Appendix B to 10 CFR Part 50 for the MSIV leakage pathway and the main condenser.

Turbine Building Design: As discussed in Section 3.1.1 of Chapter 13 of this report, the turbine building should be seismically analyzed for the SSE to ensure that it will not adversely affect the structural integrity of the main steam piping, its associated piping, and the main condenser. The staff requires that plant-specific walkdowns be conducted before commercial operation to assess the potential failures of non-seismically designed systems,

structures, and components overhead, adjacent to, and attached to the main steam piping, its associated piping, and the condenser. For the walkdowns, an appropriate seismic experience data base should be used for identifying potential failure modes in order to provide high confidence that the main steam piping, its associated piping, and the main condenser in ALWRs can retain their structural integrity during and following an SSE.

Conclusion

The approach described above for ALWRs to resolve the BWR main steamline issue provides reasonable assurance that the main steam piping from the outermost isolation valve up to the turbine stop valve, the MSIV leakage pathway (i.e., the drain line or bypass line) up to the condenser, and the main condenser will remain structurally intact, so that they can act as a holdup volume for fission products during and following an SSE. The final closure of the BWR main steamline issue will be verified through plant-specific walkdowns during the combined license phase. Although the staff has identified its position on this matter in the draft Commission paper issued on February 27, 1992, its approach to resolving this issue has not been reviewed by the Commission and, therefore, does not represent an agency position. This is an open issue that must be satisfactorily resolved before the staff can complete its review of the Evolutionary Requirements Document. This issue is also discussed under Item II.E of the draft Commission paper of February 27, 1992, contained in Annex C of this appendix.

Plateout

Plateout of radioactive iodine on the main steam pipe and condenser surfaces can result in significant dose mitigation. Several technical references indicate that particulate and elemental iodines would be expected to deposit on surfaces, and the rates of deposition would vary with temperature, pressure, gas composition, surface material, and particulate size. The staff is evaluating the extent to which credit for the fission product attenuation in the main steamlines and in the isolation condenser is appropriate and reasonable for BWRs even though the main steamlines downstream from the MSIV and its condenser are not designed to withstand the SSE as defined in Section III.C of Appendix A to 10 CFR Part 100. Since product attenuation in the main steamline and in the isolation condenser depends on the design of the plant, the staff will evaluate this issue during its review of an individual application for FDA/DC.

2.3.2 Simplification of Postaccident Sampling System

The regulations and guidance regarding the design of the postaccident sampling system (PASS) are given in 10 CFR 50.34(f)(2)(viii), RG 1.97 ("Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident"), and NUREG-0737 ("Clarification of TMI Action Plant Requirements"). Section 2.3.2 of Appendix B to Chapter 1 of the Evolutionary Requirements Document gives EPRI's proposed deviations from the design requirements for the PASS in several areas. The staff's evaluation of this issue is provided under Item II.I of the draft Commission paper dated February 27, 1992, contained in Annex C of this appendix. Although the staff has identified its position on this matter in the draft Commission paper, its approach to resolving this issue has not been reviewed

by the Commission and, therefore, does not represent an agency position. This is an open issue that must be satisfactorily resolved before the staff can complete its review of the Evolutionary Requirements Document.

2.4 Issues Related to Chapter 4 of the Evolutionary Requirements Document

2.4.1 Reactor Pressure Vessel Level Instrumentation for PWRs

Early versions of Section 2.4.1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document proposed elimination of the reactor pressure vessel level instrumentation system (RVLIS) for a PWR on the basis of the design features of an ALWR on which the Evolutionary Requirements Document is applied. Section 3.5.2.1 of Chapter 3 required that instrumentation to monitor adequate core cooling and the conditions following an accident be provided and indicates that the ALWR is intended to be consistent with the applicable regulatory guidance for monitoring postaccident conditions. However, Section 3.5.2 of Chapter 3 did not list the RVLIS as a part of monitoring capability.

As a resolution of Three Mile Island (TMI) Action Plan Item II.F.2 (NUREG-0737), 10 CFR 50.34(f)(2)(xviii) requires that commercial nuclear power plants 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 its evaluation of 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 inadequate core cooling conditions.

The staff reviewed EPRI's bases and justification for not including a requirement for a RVLIS for the ALWR and determined that EPRI did not present a sufficiently new argument to alter the NRC position requiring the RVLIS for existing PWRs.

The larger pressurizer and dedicated vent system proposed for the evolutionary plant will reduce the probability of loss of pressurizer level and 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 a RVLIS, the staff concludes that the Evolutionary Requirements Document does not meet the NRC requirement to have a RVLIS.

The objective of the policies established by NUREG-0737 and RG 1.97 and of the industry's development efforts was that reliable and unambiguous indication of inadequate core cooling (ICC) be provided. The staff recognizes that the systems developed in response to the need and to these guidance documents are not ideal; nonetheless, they have been judged to satisfy the above objective 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, EPRI should consider requiring better system performance in its Evolutionary Requirements Document. Under most conditions, the RVLIS does provide the unambiguous indication sought. The staff concludes 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 early versions of Section 2.4.1 of Appendix B to Chapter 1 of the Evolutionary 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 reactor coolant system (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 provide an adequate representation of RCS inventory. Indication was available via the RVLIS portion of the integrated core cooling monitoring system. Even though the RVLIS had not been declared operational and operator action to terminate the condition was not taken 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 identified other events during which the RVLIS (if it had been installed) could have resulted in better operator response. These events included a natural circulation cooldown event at St. Lucie Unit 1, during which a bubble had formed in the reactor vessel head, even though there was continued indication of adequate subcooling margin.

In addition to the staff requirements for vessel level instrumentation as part of an ICC detection system, such instrumentation provides 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 possibly arresting core-melt progression before vessel failure. Therefore, the staff concludes that all ALWR designs must have a RVLIS.

In Revision 4, EPRI modified Section 6.3.3.2 of Chapter 4 of the Evolutionary Requirements Document to require a RVLIS and proposed general requirements for such a system. The staff concludes that these requirements are acceptable. However, because little specific information on the RVLIS system was provided, the staff requires that each PWR ALWR designer identify system design and performance criteria including the system's potential accident-management role and the resulting severe environment to which it may be subjected. The staff will evaluate the specific design during its review of an individual application for FDA/DC.

2.5 Issues Related to Chapter 5 of the Evolutionary Requirements Document

2.5.1 Containment Leak Rate Testing

Section 2.5.1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document proposes that the maximum interval between Type C leak rate tests be 30 months instead of the 24 months maximum currently required in Appendix J to 10 CFR Part 50. EPRI originally proposed this optimization subject in Section C.1 of Revision 0 of Chapter 5 of the Evolutionary Requirements Document.

EPRI states that ALWRs will have a refueling cycle of 24 months. Therefore, it believes that a Type C test interval of at least 30 months is needed to avoid the necessity of shutting down the facility solely to perform Type C local leak rate tests. Otherwise, it may be necessary to subject the facility to the increased risks, and personnel to the increased exposure, associated with testing at power. EPRI cites the economic benefits of a 24-month refueling cycle, the reduced occupational radiation exposure associated with longer test intervals, and the relatively minor overall effect on risk should the longer test interval result in increased containment leakage. In the latter regard, EPRI cites NUREG/CR-3539, "Impact of Containment Building Leakage on LWR Accident Risk," which states that an increase in containment leakage from 0.1 percent to 1.0 percent every 24 hours would increase the overall risk by 1.5 percent.

In addition, in its letter dated March 3, 1992, EPRI provided hand-marked page changes requesting additional changes to the containment leak rate testing requirements of Appendix J to 10 CFR Part 50 as contained in ANSI/ANS 56.8, "Containment System Leakage Testing Requirements," such as relief from the test interval for air locks. The staff's evaluation of these issues is given in this report under Item II.H of the draft Commission paper dated February 27, 1992, contained in Annex C of this appendix and Section 6.3 of Chapter 5. Although the staff has identified its position on the issue of the 24-month Type C test interval in the draft Commission paper, its approach to resolving this issue has not been reviewed by the Commission and, therefore, does not represent an agency position. This is an open issue that must be satisfactorily resolved before the staff can complete its review of the Evolutionary Requirements Document.

2.5.2 Physically Based Source Term

General Discussion

Section 2.4.1.2 of Chapter 1 of the Evolutionary Requirements Document requires the use of a more realistic severe reactor accident source term for the offsite radiological consequence assessment than the current source term specified in U.S. Atomic Energy Commission report TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," March 23, 1962, and in RGs 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," and 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors." Section 2.5.2 of Appendix B to Chapter 1 lists source term treatment for the evolutionary ALWR as a plant optimization subject.

In addition to the discussion in these sections, EPRI submitted to the NRC two technical reports entitled "Evolutionary Plant Source Term Report (October 1990)" and "Passive ALWR Source Term (February 1991)." In these sections and reports, EPRI points out that ALWR licensing-design-basis requirements as well as design enhancements related to severe accidents should be based on the full body of current knowledge regarding accident source terms. EPRI believes that the evolutionary designs should be evaluated on the basis of a realistic treatment of the fission product source term, including the extensive research that has been done on fission product behavior since TID-14844 was issued, and especially since the accident at Three Mile Island Unit 2 (TMI-2) in 1979. EPRI's view is that this approach will result in designs that are improved and provide enhanced safety protection.

In SECY-89-341, "Updated Light Water Reactor Source Term Methodology and Potential Regulatory Applications," dated November 6, 1989, SECY-90-016, and SECY-90-341, "Staff Study on Source Term Update and Decoupling Siting From Design," dated October 4, 1990, the staff discussed an integrated set of its activities involving regulatory development and implementation of updated source term information in connection with its review of the Evolutionary Requirements Document and the evolutionary ALWR designs. In these papers, the staff also discussed revisions of 10 CFR Parts 100 and 50 that will decouple reactor siting from plant design.

Both the NRC and the industry have expended a great deal of effort to obtain a better understanding of the fission product transport and release mechanism following a severe reactor accident. The accident source terms proposed by EPRI are based on a single, enveloping value for a bounding severe reactor accident sequence, using release data obtained from the severe fuel damage tests at the Power Burst Facility, source term measurements at the Loss of Fluid Test (LOFT) Facility, and data from the TMI-2 postaccident examination. The NRC source term developed by the staff is based primarily on source term calculations performed by the Source Term Code Package (STCP) for individual accident sequences selected in NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants." The STCP has been demonstrated and has emerged as an integral tool for the analysis of the fission product transport and release mechanism in the reactor coolant system and in the primary containment in ALWRs under severe reactor accident conditions.

The staff finds that there is good agreement between EPRI's proposed source terms for the passive plants and those of the staff in most areas, except for the magnitude of nuclides with low volatility, where the staff's values are about an order of magnitude higher than those proposed by EPRI. However, there is less agreement between EPRI's proposed source term for the evolutionary plants and those of the staff, because EPRI's evolutionary plant source term is based on a partial core melt, whereas the staff's proposed source terms as well as those for EPRI's passive plants are based on a complete core melt. In either case, the staff concludes that the differences between the staff's and EPRI's estimates for the nuclides with low volatility will not have a major effect on offsite dose assessment. In SECY-92-127, "Revised Accident Source Terms for Light Water Nuclear Power Plants," the staff proposed a revised severe accident source term in a draft NUREG report, "In-Containment Accident Source Terms for Light Water Nuclear Power Plants." The draft NUREG report provided the types, timing, and quantities of fission products that would be released into the containment on the basis of a range of core-melt-accident scenarios, including failure of the reactor pressure

vessel and subsequent core-concrete interactions. After SECY-92-127 was written, the staff issued NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," (June 1992) for a 90-day public comment period. The staff will evaluate the entire source term-related issue during its review of the Requirements Document and individual applications for FDA/DC. This is an open issue that must be satisfactorily resolved before the staff can complete its review of the Evolutionary Requirements Document. See Item I.B of SECY-90-016 contained in Annex A and Item I.A of the draft Commission paper dated February 27, 1992, contained in Annex C of this appendix for additional discussion of this issue.

Specific Source Term Issues

In Section C.2 of the DSER for Chapter 5, the staff discussed several specific issues related to the source term. The following is the staff's evaluation of these issues.

Deletion of PWR Containment Spray Additive

In Section 1.2.3.1 of Chapter 5 of the Evolutionary Requirements Document, EPRI notes that previous regulatory practice called for a chemical additive in PWR containment spray solutions to raise the pH of the spray solution to enhance the solubility of elemental iodine in the spray droplets. EPRI also states that recent research shows that the vast majority of volatile fission products released will be in particulate form and that the containment spray additive is unnecessary and should be deleted. Also, EPRI believes that rates for fission product removal by the spray should be established on the basis of current research.

The review of the PWR spray system as a fission product removal system is covered in SRP Section 6.5.2, "Containment Spray as a Fission Product Cleanup System." The staff issued a revision of SRP Section 6.5.2 in December 1988, in which it acknowledges that a chemical additive is not necessarily required during spray injection, but that pH control should be maintained for the sump solution during postaccident conditions. The deletion of a spray additive for the ALWR is consistent with the revised SRP and is acceptable. Revised SRP Section 6.5.2 provides fission product cleanup models that can be used with either RG 1.4 or current best-estimate fission product releases.

The Evolutionary Requirements Document does not specify chemical additives for long-term containment sump pH control. This feature is normally provided by baskets of trisodium phosphate suspended in the sump. Although the proposed design of the ALWR does not include a recirculation sump, chemicals could be readily located in other locations where they would be dissolved by containment spray. In Section C.2.1 of the DSER for Chapter 5, the staff stated that EPRI should address Branch Technical Position (BTP) MTEB 6-1 (SRP Section 6.1.1, "Emergency Safety Features Materials"), which requires containment sump pH control, and identified this as an open issue. The staff will evaluate this matter during its review of an individual application for FDA/DC to ensure that the designer acceptably addresses BTP MTEB 6-1. On the basis of the above discussion, this DSER open issue is closed. The staff's evaluation of this issue can be found in Section 8.2 of Chapter 5 of this report.

Revised SRP Section 6.5.2 states that for sump solutions having a pH less than 7, molecular iodine should be conservatively assumed to evolve into the containment atmosphere. In Section C.2.1 of the DSER for Chapter 5, the staff stated that EPRI should include a requirement that the ALWR fission product cleanup analyses reflect the effect of reduced pH consistent with the methodology in revised SRP Section 6.5.2 and identified this as an open issue. The staff will evaluate this matter during its review of an individual application for FDA/DC to ensure that the effect of reduced pH is properly reflected in the cleanup analysis. On the basis of the above discussion, this DSER open issue is closed.

Deletion of Charcoal Adsorbers

In Section 1.2.3.2 of Chapter 5 of the Evolutionary Requirements Document, EPRI states that activated charcoal filters in BWR standby gas treatment systems and other PWR ventilation systems are required for many areas in which fission products could be released. It also states that these filters are required solely for removing organic and elemental iodine, are complex, and have been difficult to operate and test. EPRI states that recent research shows that the amount of elemental iodine expected to be released in an accident is small enough that charcoal filtration is unnecessary, and that the reference to charcoal filters has been deleted in the Evolutionary Requirements Document with regard to fission product filtration systems.

In SECY-86-76, the staff identified air filtration systems as an area of potential change based on source term research. Recent research indicates that the fraction of fission product iodine present in elemental and organic forms may be lower than that specified in present regulatory guidance (RGs 1.3 and 1.4), but may still be at a level at which some charcoal filtration is warranted. Further, this research also indicates that the quantity of airborne material (radioactive and nonradioactive aerosols) expected to be produced in accident sequences of interest could be large in comparison to the retention capacity of the high-efficiency particulate air filters in present engineered safety features air filtration systems. Finally, many filter systems are also used to process activity released during normal operation during which charcoal may be useful. The staff intends to consider potential changes in air filtration systems based on the proposed revised source term discussed in SECY-92-127, but concludes that the complete removal of charcoal in air filtration systems, as stated in EPRI's position, is currently unjustified. In particular, the staff notes that there are large uncertainties with regard to gaseous iodine production due to pH changes in the in-containment refueling water storage tank, hydrogen burns, and irradiation of other chemical forms of iodine. Therefore, in Section C.2.2 of the DSER for Chapter 5, the staff concluded that the need for activated charcoal filters in appropriate ALWR ventilation systems must be evaluated as part of the overall development of the updated analytical methodology for the source term to be used for ALWRs and identified this as an open issue. The staff will evaluate this matter during its review of an individual application for FDA/DC. On the basis of the above discussion, this DSER open issue is closed.

BWR Suppression Pool Fission Product Scrubbing

In Sections 1.2.3.3 and 7.3.11 of Chapter 5, EPRI states that previous licensing practice did not recognize scrubbing of fission products by a BWR suppression pool and that recent research has shown such scrubbing to be

effective. EPRI originally took the position that a decontamination factor for radionuclide scrubbing by BWR suppression pools should be credited on the basis of evaluations of accident sequences using the MAAP code and decontamination factors determined by SUPRA, SPARC, or other technically defensible methods.

RG 1.3 (Position C.1.f) allows no credit for retention of iodine by a BWR suppression pool. However, as discussed in Section C.2.3 of the DSER for Chapter 5, the staff has issued a revised SRP Section 6.5.5, "Pressure Suppression Pools as Fission Product Cleanup Systems," dated December 1988. The essential feature of this revised section is that it recognizes that suppression pools are capable of scrubbing non-noble gas fission products and it would be an undue conservatism to ignore this capability. On the basis of the revised SRP Section 6.5.5, the staff concluded that credit may be given for suppression pool fission product removal, provided suppression pool decontamination factors are evaluated in accordance with the methodology prescribed in the revised SRP section and identified this as an open issue.

The staff concludes that this criterion is reflected in Table B.1-2 of Appendix B to Chapter 1 of the Evolutionary Requirements Document. In addition, Section 1.2.3.3 of Revision 1 of Chapter 5 has been modified to delete the reference to specific alternative methodologies. A statement to use SRP Section 6.5.5 methodologies was added to Section 7.3.11 of Chapter 5. These changes are consistent with the staff's position and are acceptable. Therefore, this DSER open issue is closed.

Timing of Fission Product Release

In current licensing practice, the assumption is that design-basis-accident fission product releases to the containment atmosphere occur virtually instantaneously. EPRI claims this assumption results in closure times of containment isolation valves that are shorter than necessary and capacities of ventilation fans that are larger than necessary, on the basis of analysis of sequences expected to dominate the likelihood of core damage. In Section 1.2.3.4 of Chapter 5 of the Evolutionary Requirements Document, EPRI takes the position that an accidental release of fission products into the containment should be assumed to occur no sooner than about 1 hour after reactor scram.

The basis for this assumption is given in EPRI's technical reports entitled "Evolutionary Plant Source Term Report (October 1990)" and "Passive ALWR Source Term (February 1991)." These reports state that the initial fission product release from the reactor coolant system to the containment atmosphere (gap activity) occurs more than 30 minutes after scram for existing plants. The reports further state that allowing for the enhancements provided by the ALWR design requirements, significant radioactive release (fuel activity) to the containment will occur no sooner than about 60 minutes after reactor scram for a plant designed to meet the Evolutionary Requirements Document.

For licensing purposes, timing of accident fission product releases into the containment is given by Positions C.1.a of RGs 1.3 and 1.4. These positions indicate that fission product releases should be assumed to be "immediately" available for leakage from the containment. In practice, the staff has typically taken this to mean "within 15 seconds" in order to allow for closure of containment isolation valves. The assumption of the release of the

quantity of fission products contemplated by RGs 1.3 and 1.4 within about 15 seconds is generally recognized as highly conservative for many accident sequences. This timing appears to be dependent on the accident sequence as well as the reactor type. Furthermore, the staff has not, at this time, decided whether certain sequences should be excluded from consideration because they are unlikely to dominate the overall likelihood of core damage. For these reasons, the staff believes that some relaxation in the assumed timing of large fission product releases into the containment associated with fuel degradation and melting may be warranted, but cannot support the specific value (1 hour after scram) proposed in the Evolutionary Requirements Document. In SECY-92-127, the staff indicated that the revised source term may make it possible to relax the closure time of certain containment isolation valves on the basis of the longer time interval before fuel failure. It may also make it possible for plants with secondary containments (e.g., BWRs) to relax the time needed to attain a negative pressure within the secondary containment because initial fission product releases from the primary containment will be relatively small. The staff regards the issue of timing of the fission product release to be part of the overall open issue regarding the source term. The staff will evaluate this matter during its review of the Requirements Document and individual applications for FDA/DC.

2.5.3 Dedicated Containment Vent Penetration

In Section 2.5.3 of Appendix B to Chapter 1, EPRI states that it has provided a criterion for protecting the containment from long-term overpressurization solely through the containment design or, alternatively, the use of pressure relief.

In SECY-90-016, the staff recommended that the Commission approve the use of an overpressure protection system that uses a dedicated containment vent for the Advanced BWR (ABWR) to avoid gross containment failure resulting from postulated slow-rising-overpressure scenarios that could result from postulated multiple safety system failures. In its staff requirements memorandum of June 26, 1990, the Commission approved the use of the containment overpressure protection system for the ABWR, subject to a comprehensive regulatory review to weigh the "downside" risks with the mitigation benefits of the system. In addition, the Commission directed the staff to ensure that full capability to maintain control over the venting process is provided in the design.

In Section 6.6.2.6 of Chapter 5 of the Evolutionary Requirements Document, EPRI provides requirements for protecting the containment from long-term overpressurization through the containment design or, alternatively, the use of pressure relief. The staff concludes that these requirements are consistent with the above guidelines and are acceptable. However, because the Evolutionary Requirements Document does not provide detailed design requirements, the staff will evaluate this matter during its review of an individual application for FDA/DC.

2.5.4 Anticipated Transients Without Scram

In Section 2.5.4 of Appendix B to Chapter 1 of Revision 1 of the Evolutionary Requirements Document, EPRI stated that the Evolutionary Requirements Document fully addressed the regulatory requirements of 10 CFR 50.62 and General Design Criterion 26 in Appendix A to 10 CFR Part 50 regarding reduction of the

probability of an anticipated-transient-without-scram (ATWS) event and enhancement of the mitigation capability should such an event occur in the evolutionary BWR. EPRI stated that, because of the increased margins, reduced demands on the reactor protection system, and the enhanced reliability of the control rod drive system, automatic initiation of the standby liquid control system (SLCS) as required by 10 CFR 50.62 was not necessary for the evolutionary BWR.

However, in its December 6, 1991 letter, EPRI stated that it had determined that automatic actuation of the SLCS was appropriate for the evolutionary BWR design. In Revision 4, EPRI deleted Section 2.5.4 from Appendix B to Chapter 1 of the Evolutionary Requirements Document because it no longer considers this to be a plant optimization subject for the evolutionary plant design. In addition, EPRI modified Section 4.6.3 of Chapter 5 to require automatic initiation of the SLCS.

The staff concludes that these changes are in accordance with 10 CFR 50.62 and are, therefore, acceptable. Therefore, this issue is closed. This issue is also discussed in this report under Item II.A of SECY-90-016 contained in Annex A of this appendix and Section 4.2 of Chapter 5.

3 GENERIC SAFETY ISSUES

3.1 Introduction

In the DSER for the various chapters of the Evolutionary Requirements Document, the staff addressed 62 generic safety issues. Table 3B.1 in this appendix lists these issues and gives the status of each as provided in the appropriate DSER chapter. The table shows that 29 of these issues remained open as a result of the staff's review at the DSER stage and 3 were confirmatory issues. Eleven issues had vendor items. The staff's final evaluation of these issues is provided in Section 3.2.

In Section 3.1 of Appendix B to Chapter 1 of the current version of the Evolutionary Requirements Document, EPRI addresses 24 generic safety issues for which there was no generic resolution as of January 1990 in the NRC's generic issues management control system (GIMCS) and which EPRI determined were technically relevant to the ALWR evolutionary plant design using the criteria in NUREG-1197, "Advanced Light Water Reactor Program - Program Management and Staff Review Methodology." These issues* are the following:

I.D.3	29	84	128
B-17	57	87	130
B-56	70	94	135
C-8	75	105	HF 4.4
15	79	113	HF 5.1
23	83	121	HF 5.2

Seven of these issues (I.D.3, B-56, 15, 57, 83, 87, and 135) were not addressed in the staff's DSERs; since January 1990, eight (29, 70, 75, 79, 84, 87, 94, and 135) have been technically resolved by the staff. The staff will evaluate the proposed elements for resolving these issues on the basis of the final agency issue resolution.

A note to Table B.3-1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document states that the screening criteria of NUREG-1197 were used to determine that Issues I.D.5(3), II.H.2, II.J.4.1, B-55, B-61, B-64, 106, and HF 4.4 were not technically relevant to the ALWR design. The Evolutionary Requirements Document, however, does not give the reasons why EPRI determined specific issues were not technically relevant to the ALWR design. The inclusion of Issue HF 4.4 in this note, for instance, appears to be in error in view of the fact that the Evolutionary Requirements Document proposes elements of resolution that include this issue. In addition, although the document does not propose elements for resolving Issue II.J.4.1, it includes commitments to comply with the revised regulations that form the basis for the staff's resolution of the issue.

To confirm and update EPRI's list, the staff used Appendix B, "Applicability of NUREG-0933 Issues to Operating and Future Plants," to NUREG-0933, "A Prioritization of Generic Safety Issues." The staff supplemented the information in Appendix B to NUREG-0933 with status information on various safety issues contained in the first-quarter fiscal year 1992 update of the

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*The issue titles and status are given in Section 3.2.

GIMCS report (memorandum for J. Taylor from E. Beckjord dated March 11, 1992). In confirming and updating EPRI's list, the staff identified 28 unresolved issues in Appendix B to NUREG-0933 as of December 31, 1991. Of these 28, for various reasons, five issues (I.D.5(3), II.J.4.1, B-55, B-61, and B-64) are not applicable to the ALWR Program. These five issues are also identified in the note to Table B.3-1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document as noted in the preceding paragraph and are discussed by the staff in Sections 3.2.2, 3.2.5, 3.2.13, 3.2.15, and 3.2.16 of this appendix. During its review, the staff noted that the current version of Appendix B to NUREG-0933 may not properly include all the generic safety issues applicable to future plants. The staff is evaluating the applicability of these issues to future designs and may need to modify Appendix B to NUREG-0933 as a result of this review.

Two issues (C-8 and 84) identified in the Evolutionary Requirements Document are not included in Appendix B to NUREG-0933. These issues were included in Appendix B to NUREG-0933 as of December 31, 1989, but were later dropped. They are discussed in Sections 3.2.17 and 3.2.32, respectively, of this appendix. Issue 135, which was resolved by the staff in March 1991, is not included in Appendix B to NUREG-0933. However, since proposed elements of resolution are provided in Appendix B to Chapter 1 of the Evolutionary Requirements Document, it is evaluated in Section 3.2.59.

In Section 3.2 of this appendix, the staff provides either an evaluation of each of these issues and the status of the staff's review or an explanation as to why the issue is not specifically evaluated. Since the generic agency resolution has not yet been identified for many of the issues, the staff may have to reevaluate the FDA/DC applicant's proposed resolution of individual issues once the generic resolution is determined. Where an issue's elements of resolution as proposed in the Evolutionary Requirements Document do not completely address the generic agency resolution of an issue or where the staff has postponed its evaluation of an issue until it reviews individual applications, the staff expects the applications for FDA/DC to address the final resolution in accordance with 10 CFR 52.47(a)(1)(iv) for the staff's review.

In its review, the staff identified 40 issues that are considered to be applicable to future plants (since they are listed in Appendix B to NUREG-0933), but that have not yet been prioritized. These are identified by number and title in Table 3B.2 of this appendix. Although five of these issues (2, 76, 110, 118, and 132) have not been prioritized, the staff evaluated each in the DSER. In addition, as of December 31, 1991, the staff has identified six new issues, which are identified by number and title in Table 3B.3. However, it has not received specific information on these issues or whether it has been determined that they are applicable to future plants.

Although the staff is not considering the unprioritized and new issues at this time, it will monitor their safety status. The vendors will be required to address, in accordance with 10 CFR 52.47(a)(1)(iv), any that are applicable to the ALWR evolutionary plant designs and that are prioritized as high or medium. For the five unprioritized issues that the staff evaluated in the DSER and that contained open or confirmatory issues, the staff will defer their further evaluation to its review of applications for FDA/DC.

Irrespective of the specific issues addressed in this evaluation and in the Evolutionary Requirements Document, FDA/DC applications are required by 10 CFR 52.47(a)(1)(iv) to contain proposed technical resolutions of those unresolved safety issues and high- or medium-priority generic safety issues that are identified in the version of NUREG-0933 current 6 months before the application and that are relevant to the design.

3.2 Staff Evaluations of Generic Issues

3.2.1 I.D.3, Safety System Status Monitoring

Issue: This issue concerns the performance of a study to determine the need for all licensees and applicants not committed to Regulatory Guide (RG) 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," to install a bypass and inoperable status indication system or similar system.

EPRI Proposal: Section 3.10.4.4 of Appendix B to Chapter 1 and Table B.1-2 in Appendix B to Chapter 1 of the Evolutionary Requirements Document show an ALWR Program commitment to comply with RG 1.47, Revision 0.

Staff Evaluation: The commitment to comply with RG 1.47 satisfies the staff's primary concern regarding this issue at this time. Issue I.D.3 has not yet been generically resolved by the staff and has a medium safety priority. Should final agency resolution of this issue dictate additional actions for ALWR evolutionary plant designs, the staff will expect the FDA/DC applicant to address the resolution of Issue I.D.3 for staff evaluation.

3.2.2 I.D.5(3), Improved Control Room Instrumentation Research - On-Line Reactor Surveillance System

Issue: The objective of this issue is to (1) perform research to determine the feasibility of detecting and diagnosing nuclear power plant operating anomalies using a continuous, on-line noise surveillance system; (2) demonstrate such a system in an NRC-licensed commercial BWR plant (which has been done); and (3) prepare a report summarizing the conclusions and recommendations from the research.

EPRI Proposal: In the note to Table B.3-1 in Appendix B to Chapter 1 of the Evolutionary Requirements Document, EPRI states that Issue I.D.5(3) was determined, through the NUREG-1197 screening process, to be not technically relevant to the ALWR design.

Staff Evaluation: The staff considers this issue to be a research item that has no apparent effect on ALWR evolutionary plant designs at this time. Appendix A to NUREG-1197 identifies issues that the staff determined were not applicable to the ALWR Program according to six categories. Although Issue I.D.5(3) is not specifically listed in any of the categories, the staff believes that it should be considered a Category (or Code) 2 item. Category 2 includes issues that address NRC training programs, policy developments, organizational changes, studies, code development and maintenance, and research activities not directly affecting plant design.

On the basis of its ongoing programs, the staff has identified a technical resolution for this issue. Should final agency resolution of this issue dictate that this issue is otherwise applicable to ALWR evolutionary plant designs, the staff will expect the FDA/DC applicant to address the resolution of Issue I.D.5(3) for staff evaluation.

3.2.3 II.E.4.3, Containment Design - Integrity Check

Issue: Following an outage, a plant could inadvertently be returned to operation without all access openings being closed. This issue originated with the discovery at a nuclear power plant in 1979 that two 3-inch containment exhaust valves had been left open for approximately 1½ years. To ensure containment integrity, an independent means should exist for directly verifying containment integrity during plant operation.

EPRI Proposal: The EPRI-proposed resolution of this issue is through the incorporation of design features to enhance administrative control of containment integrity. In addition to the leak rate testing specified to satisfy regulations, Chapter 5 of the Evolutionary Requirements Document states that specific design features will focus on reducing the probability of a containment penetration being inadvertently left open. By clearly identifying all isolation valves, ensuring easy access for verification, providing better position indication, and including fail-safe design features, the probability of a large penetration being left open is substantially reduced. In addition, Section 6.3.2.5 of Chapter 5 requires the means to enable the operator to perform a periodic check for gross leakage of containment atmosphere during normal operation. Instrumentation for other purposes (i.e., that used for Type A integrity leakage rate testing and air or gas addition flow monitors) will be used to the extent practicable for this function.

Staff Evaluation: In the DSER for Chapter 5, the staff reported that its assessment (NUREG-1273, "Technical Findings and Regulatory Analysis for GSI II.E.4.3, 'Containment Integrity Check,'") of Issue II.E.4.3 generically resolved the issue for operating reactors and questioned the cost-benefit of a backfit for current operating plants that would require a continuous or short-term periodic gross check of containment integrity. The staff indicated, however, that such a system would not be a backfit for an ALWR and that it was not evident, from the information in the Evolutionary Requirements Document, that evolutionary ALWRs would have a significantly reduced number of containment penetrations. The staff concluded that the next generation of LWRs should have an improved capability to detect inadvertent containment bypass and that the Evolutionary Requirements Document should address these concerns. EPRI has revised the Evolutionary Requirements Document stating that such improved detection capability will be available. Therefore, this open issue is resolved.

3.2.4 II.H.2, Obtain Technical Data on the Conditions Inside the Three Mile Island, Unit 2, Containment Structure

Issue: This issue entails the collection of certain technical information on the conditions of the facility at Three Mile Island, Unit 2, as cleanup operations proceed. The information is to be disseminated under another generic safety issue (II.H.3). The information obtained under this issue will

be used in resolving other generic safety issues, such as Issues A-45 and A-48 (former unresolved safety issues) and Issues II.B.5, II.B.7, II.B.8, and II.E.3.4.

EPRI Proposal: In the note to Table B.3-1 in Appendix B to Chapter 1 of the Evolutionary Requirements Document, EPRI states that Issue II.H.2 was determined, through the NUREG-1197 screening process, to be not technically relevant to the ALWR design.

Staff Evaluation: This issue entails the collection of information that will be used in resolving other specific generic safety issues and, therefore, is not to be analyzed separately. Also, Appendix A to NUREG-1197 identifies issues that the staff determined were not applicable to the ALWR Program according to six categories. Issue II.H.2 is listed as a Code (or Category) 5 item, that is, an issue that is not applicable to plant design. In addition, according to Appendix B to NUREG-0933, Issue II.H.2 is not applicable to future plants; therefore, an evaluation is not necessary.

3.2.5 II.J.4.1, Revise Deficiency Reporting Requirements

Issue: The event reporting requirements of 10 CFR Part 21 and the deficiency reporting requirements of 10 CFR 50.55(e) are being revised to ensure that licensees report all reportable items promptly to the NRC and that the information submitted is complete. The reports received as a result of the anticipated rule changes are expected to (1) provide increased information on component failures that affect safety (so that prompt and effective corrective action can be taken) and (2) be used as input to an augmented role of the NRC's vendor and construction inspection programs.

EPRI Proposal: In the note to Table B.3-1 in Appendix B to Chapter 1 of the Evolutionary Requirements Document, EPRI states that Issue II.J.4.1 was determined, through the NUREG-1197 screening process, to be not technically relevant to the ALWR design. This may be in error, however, since Table B.1-1 in Appendix B to Chapter 1 shows an ALWR Program commitment to comply with 10 CFR Part 21 and 50.55(e).

Staff Evaluation: The staff has identified a resolution of Issue II.J.4.1, that is, to promulgate changes to 10 CFR Part 21 and 50.55(e). Final rule changes were published in the Federal Register (56 FR 36081) on July 31, 1991. EPRI's commitments to comply with 10 CFR Part 21 and 50.55(e) satisfy the staff's primary concerns regarding this issue at this time.

The discussion in NUREG-0933 of Issue II.J.4.1 has not been updated to reflect the recent rule changes. Should the final agency resolution of this issue dictate additional actions for ALWR evolutionary plant designs, the staff will expect the FDA/DC applicant to address those actions for staff evaluation.

3.2.6 A-29, Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage

Issue: Reduction of the vulnerability of reactors to radiological sabotage is currently treated as a plant physical security function and not as a plant design requirement. Although present reactor designs provide a great deal of inherent protection against industrial sabotage, extensive physical security measures are still required to provide an acceptable level of protection. An

alternative approach would be to consider more fully reactor vulnerabilities to sabotage during the preliminary design phase. Because emphasis is being placed on standardizing plants, it is especially important to consider measures that could reduce the vulnerability of reactors to sabotage. Any design features to enhance physical protection must be consistent with system safety requirements.

EPRI Proposal: Section 1.4.1 of Chapter 9 of the Evolutionary Requirements Document states that substantial inherent resistance to sabotage will be provided in an ALWR plant through rugged, reinforced-concrete external walls and internal barriers that restrict access. It also states that further resistance will be provided through physically separated, redundant safety systems that provide a high reliability for the prevention of core damage and the mitigation of the consequences of initiating events, including sabotage. The access control features of the physical security system and the vital area barriers will provide the primary defense against the internal threat.

In addition, the Evolutionary Requirements Document indicates that the design will ensure that the necessary security functions are provided without unnecessarily constraining plant operability or impeding access to or egress from vital areas under emergency conditions.

The Evolutionary Requirements Document includes the following key requirements that are applicable to both BWRs and PWRs:

- Section 2.2.5 of Chapter 5 requires that the design features, configuration, and divisional separation of the engineered safety systems be integrated with the building design and the plant's security system to enhance the plant's capability for protection against sabotage.

- Section 2.3.2 of Chapter 5 requires that each division of safety system functions be totally independent and separated both mechanically and electrically except for areas in which it is physically impracticable or less safe. Requirements for mechanical separation that enhance protection from sabotage include 3-hour-fire barriers, physical protection of each division from flooding, and physical protection from pipe breaks both inside and outside the containment. Additionally, the physical barriers between divisions should be designed to enhance the plant's capability to resist sabotage.

- Section 1.5.2 of Chapter 6 summarizes EPRI's policy related to plant arrangement and the physical separation of engineered safety systems to reduce the vulnerability of ALWR plants to sabotage.

- Sections 2.3.5.4 and 2.3.8 of Chapter 6 provide specific requirements for plant arrangement that provide for protection from external threats and maximize security by enhancing sabotage resistance. EPRI states that the plant will be arranged to provide as much inherent protection as possible. Also, the plant designer will incorporate physical protection measures, such as hardening of vital walls, floors, and ceilings, into the design of the areas that house equipment performing vital functions. Utility ports penetrating vital area boundaries will be minimized and barriers will be designed to protect vital systems from sabotage.

- Section 5 of Chapter 9 provides the requirements for the site security system, including ensuring the compatibility of security system requirements with plant arrangement and safety system design, definition of vital systems, layout of vital components, security barriers, intrusion detection systems, isolation zone requirements, security alarms, access control, security communications, power supply, and data management.

Staff Evaluation: In its DSER for Chapter 9, the staff concluded that the Evolutionary Requirements Document had adequately addressed Issue A-29 with respect to external sabotage, but that it had not adequately addressed insider sabotage. At that time, the discussion of this issue in Appendix B to Chapter 9 of the Evolutionary Requirements Document made no reference to the potential for sabotage by a "knowledgeable insider" with authorized access or to acts of sabotage that could occur during maintenance activities. In the DSER, the staff noted that in a letter to the NRC dated August 18, 1989, EPRI had committed to expand Section 5.2.2.1 of Chapter 9 to require that ALWR plant designers analyze the vulnerability of their designs to insider sabotage before finalizing the designs. The staff has confirmed that this section of the Evolutionary Requirements Document includes a requirement for the analysis to consider both insider and outsider threats. Therefore, this confirmatory issue is resolved.

3.2.7 A-40, Seismic Design Criteria (former unresolved safety issue)

Issue: Issue A-40 was formulated to identify and quantify conservatism or nonconservatism inherent in seismic design criteria for nuclear power plants. Several years of research and coordination with the industry convinced the staff that some changes in the regulatory guidance in the Standard Review Plan (SRP, NUREG-0800) for seismic design were warranted.

As the work on Issue A-40 progressed, the staff decided to modify the regulatory guidance for seismic design in the SRP by revising the appropriate sections of the SRP to include the current state of knowledge in the licensing criteria. As a result of the resolution of Issue A-40, Revision 2 of SRP Sections 2.5.2, "Vibratory Ground Motion"; 3.7.1, "Seismic Design Parameters"; 3.7.2, "Seismic System Analysis"; 3.7.3, "Seismic Subsystem Analysis"; and 3.7.4, "Seismic Instrumentation," was issued in August 1989. All new applicants were required to comply with these revised SRP sections.

EPRI Proposal: Methods for operating-basis-earthquake (OBE) and dynamic analysis are addressed in Section 2 of Appendix B to Chapter 1 of the Evolutionary Requirements Document.

Staff Evaluation: EPRI's proposal for certain aspects of Issue A-40 differs from the staff's acceptance criteria in the SRP sections referenced above, as follows:

- The Evolutionary Requirements Document states that the requirement in Appendix A to 10 CFR Part 100 that structures, systems, and components be designed to withstand an OBE should be eliminated, and that the OBE should be decoupled from the safe shutdown earthquake (SSE). Currently, 10 CFR Part 100, Appendix A requires that the magnitude of the OBE be at least one-half that of the SSE.

Although Appendix A to 10 CFR Part 100 is being revised, decoupling the OBE from SSE is currently an exception to Appendix A to 10 CFR Part 100. If an ALWR applicant wishes to decouple the OBE from the SSE, it needs to specify the decoupling criteria, which the staff will evaluate and address as appropriate on a design-specific basis. This issue is discussed in this report in Section 4 of Chapter 1, Section 2.1.1 of Appendix B to Chapter 1, under Item IV.A of SECY-90-016 contained in Annex A of this appendix, under Item I.M of the draft Commission paper issued on February 27, 1992 in Annex C, and under Item C of the draft Commission paper issued on July 6, 1992, in Annex D.

Since the Evolutionary Requirements Document does not provide sufficient information to enable the staff to evaluate the design requirements in this area, the staff will evaluate this area during its review of individual applications for FDA/DC.

- Regarding the damping values for seismic design for the OBE and SSE given in RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," the Evolutionary Requirements Document states that since the SSE damping values are based on conservative lower bounds, higher damping values would be more realistic. Consequently, EPRI proposes to use the damping criteria of American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section III, Code Case N-411 and to delete the use of the OBE damping values.

  The Code Case N-411 damping values are meant to be used for piping and equipment and not for seismic design of structures. The staff has accepted the Code Case N-411 damping values for piping and equipment supports provided the additional guidance of RG 1.84, "Design and Fabrication Code Case Acceptability - ASME Section III, Division 1," is followed. The use of the damping values of Code Case N-411 for structures for SSE loads and the deletion of the OBE damping values of RG 1.61 is a deviation from the established staff position. Justifications for such deviations must be provided and can be accepted by the staff only when the decoupling issue discussed above is resolved.

  Since the Evolutionary Requirements Document does not provide sufficient information to enable the staff to evaluate the design requirements in this area, the staff will evaluate this area during its review of individual applications for FDA/DC.

- The Evolutionary Requirements Document states that the modal combination of high-frequency modes for vibratory loads with significant high-frequency input (33 to 100 Hz) should be done using the algebraic sum method.

  The algebraic sum method for modal combination of high-frequency modes for vibratory loads is a deviation from RG 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," and Appendix A to SRP Section 3.7.2. Either Appendix A to SRP Section 3.7.2 should be used, or additional justification for this position must be provided for staff review.

Since the Evolutionary Requirements Document does not provide sufficient information to enable the staff to evaluate the design requirements in this area, the staff will evaluate this area during its review of individual applications for FDA/DC.

- The Evolutionary Requirements Document states that the use of the spectral peak shifting technique should be permitted as an alternative to spectrum broadening.

  The use of the spectral peak shifting technique rather than spectrum broadening is a deviation from RGs 1.92 and 1.122, "Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components." The staff has accepted the spectral peak shifting technique on a case-by-case basis after evaluating the frequency content and proximity of the peaks for plant-specific designs. Therefore, the staff will evaluate this exception during its review of individual applications for FDA/DC.

3.2.8 A-46, Seismic Qualification of Equipment in Operating Plants (former unresolved safety issue)

Issue: The design criteria and methods for the seismic qualification of mechanical and electrical equipment in nuclear power plants have undergone significant change during the commercial nuclear power program. Consequently, the margins of safety in existing equipment to resist seismically induced loads and perform the intended safety functions may vary considerably. The seismic qualification of the equipment in operating plants must, therefore, be reassessed to ensure the capability to bring the plant to a safe shutdown condition after a seismic event. The objective of this issue was to establish an explicit set of guidelines that could be used to judge the adequacy of the seismic qualification of mechanical and electrical equipment at all operating plants in lieu of attempting to backfit current design criteria for new plants.

EPRI Proposal: Section 12.6.2 of Chapter 1 of the Evolutionary Requirements Document requires that anchor bolts for flat-bottomed tanks be extended at least 2 feet above the concrete foundation level and that the anchor bolt chairs be extended full height and continuously welded to the tank wall. These requirements are intended to distribute seismic forces and to develop ductility.

Staff Evaluation: In the DSER for Chapter 6, the staff stated that the Maine Yankee Seismic Margin Study and the resolution of Issue A-46 showed that the anchorage of the safety-related tanks was a weakness at operating plants and concluded that EPRI should emphasize tank anchorage design and installation. The staff has reviewed the requirements in Section 12.6.2 of Chapter 1 of the Evolutionary Requirements Document pertaining to the extension of anchor bolts and anchor bolt chairs for flat-bottomed tanks and the welding of anchor bolt chairs to tank walls and concludes that they adequately address the DSER open issue regarding tank anchorage design and installation. Therefore, this open issue is resolved.

3.2.9 A-47, Safety Implications of Control Systems (former unresolved safety issue)

Issue: Concerns have been raised regarding the potential for accidents or transients being made more severe as a result of control system failures, including control and instrumentation power supply faults. During the licensing review process, the staff performs an audit review of the non-safety-grade control systems to ensure that an adequate degree of separation and independence is provided between these non-safety-grade systems and the safety systems. On this basis, it is generally believed that control system failures are not likely to result in loss of safety functions that could lead to serious events or result in conditions that the safety systems are not able to mitigate. However, indepth studies for all non-safety-grade systems have not been performed.

EPRI Proposal: Table B.1-2 in Appendix B to Chapter 1 of the Evolutionary Requirements Document shows an ALWR Program commitment to comply with Generic Letter (GL) 89-19, "Request for Action Related to Resolution of Unresolved Safety Issue A-47, 'Safety Implication of Control Systems in LWR Nuclear Power Plants.'"

Staff Evaluation: The staff has resolved Issue A-47 and has issued GL 89-19. GL 89-19 contains several recommendations regarding protection from certain control system failures and modification of selected emergency procedures to ensure that plant transients resulting from control system failures do not compromise public safety.

In its DSER for Chapter 10, the staff concluded that EPRI's commitment (letter to the NRC dated December 21, 1990) to include a requirement that the ALWR designer comply with GL 89-19 was acceptable and that it would confirm that this commitment was acceptably incorporated into the Evolutionary Requirements Document. The staff has confirmed that Table B.1-2 in Appendix B to Chapter 1 indicates a commitment to comply with GL 89-19. Therefore, this confirmatory issue is resolved.

The staff also concluded that the Evolutionary Requirements Document did not provide the necessary design details to be able to determine if the plant-specific design and arrangement will be adequate to resolve this issue. Therefore, the staff will evaluate resolution of Issue A-47, including aspects of digital control and safety systems, during its review of individual applications for FDA/DC. See also the evaluation of Issue 76 in Section 3.2.28.

3.2.10 A-48, Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

Issue: As a result of the accident at Three Mile Island, Unit 2 (TMI-2), the Commission promulgated regulatory requirements on hydrogen control in 10 CFR 50.34 and 50.44. 10 CFR 50.34(f) requires a hydrogen control system based on a 100- percent fuel-cladding metal-water reaction and a hydrogen concentration limit of 10 percent on uniformly distributed hydrogen in the containment or a postaccident atmosphere that will not support hydrogen combustion. Plants covered by this rule included only those whose construction permits had not been issued at the time of the TMI-2 accident. The issue is whether this requirement applies to the evolutionary plants.

EPRI Proposal: In Section 3.5.3 of Appendix B to Chapter 1 of the Evolutionary Requirements Document, EPRI proposes that the evolutionary plant design have the capability to ensure that necessary accident prevention and mitigation functions can be performed during and after events in which hydrogen is produced. The Evolutionary Requirements Document requires the design to accommodate an amount of hydrogen equivalent to that generated by oxidation of 75 percent of the fuel cladding surrounding the active fuel. This is to be accomplished for a PWR dry containment by containment volume and mixing so that the uniformly distributed hydrogen gas concentration in the containment does not exceed 13 percent under dry conditions. For a BWR, the Evolutionary Requirements Document permits provision of a noncombustible containment atmosphere as an acceptable alternative approach.

Staff Evaluation: In Appendix B of its DSER for Chapter 5, the staff concluded that EPRI's proposed approach to resolving this issue was unacceptable. On the basis of the proposed resolution now in the Evolutionary Requirements Document, the staff agrees with EPRI's proposal on hydrogen control for BWRs, which indicates that inerting is an acceptable approach. This will meet the requirements of 10 CFR 50.44, the issuance of which resolved Issue A-48.

In SECY-90-016, the staff recommended to the Commission that the hydrogen control requirements for evolutionary plants be identical to those stated in 10 CFR 50.34(f). The Commission approved the staff's recommendation in a staff requirements memorandum dated June 26, 1990. In its letter to the NRC dated December 6, 1991, EPRI stated that the Evolutionary Requirements Document would be modified to fully comply with the staff position of 100-percent active fuel cladding and a maximum containment hydrogen gas concentration of 10 percent. This commitment is acceptable and resolves the open issue in the DSER. However, since the staff position in SECY-90-016 is that the plant-specific design must comply with 10 CFR 50.34(f) for combustible gas control, the staff will review relevant design features for conformance to 10 CFR 50.34(f) during its review of individual applications for FDA/DC. See also the evaluations in Sections 2.3 and 6.5.1 of Chapter 5 of this report and the evaluation of Issue 121 in Section 3.2.46 of this appendix.

3.2.11 B-17, Criteria for Safety-Related Operator Actions

Issue: Issue B-17 entails the development of criteria for safety-related operator action (SROA) during the response to or recovery from transients and accidents. The criteria would include a determination of actions that are to be automated in lieu of operator action and the development of a time criterion for SROA. Of particular concern is consideration of the need to automate emergency core cooling system (ECCS) realignment following a loss-of-coolant accident (LOCA), which currently requires manual operation in some PWRs. Such a realignment is dependent on break size and must be done before the borated water storage tank is depleted. Implementation of this criterion may result in changes and additions to the design of the engineered safety features control systems.

EPRI Proposal: Section 3.10.2.4.1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document states that the ALWR plant design requires automation of the initiation of protection and safety systems and that the automation of the operation of these systems precludes the need for operator

action for at least 30 minutes after the initiation of an event. The capability for manual initiation and operation is not precluded, and adequate information and control capabilities are required so that operators can effectively provide backup for automated actions.

Section 3.10.2.4.2 of Appendix B to Chapter 1 states:

- Sections 2.3.2.9, 2.3.3.5, and several places in Table 1.2-5 of Chapter 1 of the ALWR Requirements Document specify the ALWR overall requirements for the time after an event is initiated in which the operator must act as not less than 30 minutes, assuming a single failure.

- Section 2.1.1 of Chapter 10 states the objective of the ALWR design shall be to take full advantage of operator capabilities, but not to challenge operator limitations.

- Section 2.2.9 of Chapter 10 states the policy for the design of the ALWR M-MIS [man-machine interface system] which requires that each monitoring, control, and protection function be evaluated as part of the design process to determine the appropriate level of automation. Consideration is required of such factors as operator workload, system response, operation complexity, level and duration of the operator's attention, and failure of the automatic features.

- Section 3.4.3 of Chapter 10 requires that the design choice on automatic versus manual control or monitoring is to be based on evaluations which specifically include consideration of operator workload, operator capability, past experience with automatic or manual controls or monitoring in similar applications, operator vigilance and the need to keep the operator involved and knowledgeable as to the plant status, amount and complexity of M-MIS equipment (including software) and the resulting maintenance and testing burden, the consequences of and potential for malfunctions of the automatic equipment and for operating errors, and regulatory requirements.

- Section 8.2.3 of Chapter 10 defines the control and monitoring strategies which shall be used for protection and safety systems. This includes the requirement that startup or actuation of these systems shall be automatic but with an effective manual backup. It also requires that these systems operate automatically after actuation for at least 20 minutes and that manual operations be limited so that essentially continuous manning for extended periods of time, e.g., hours or days, is not required.

- The part of this issue concerning automatic versus manual ECCS realignment does not apply to the ALWR PWR. The PWR safety injection system (SIS) maintains inventory by pumping water from an in-containment refueling water storage tank (IRWST) to the reactor vessel. The IRWST provides a continuous source of water to the SIS pumps because water collects in the IRWST and thus eliminates the need for realignment of the SIS pumps for long-term post-LOCA recirculation.

Section 3.10.2.4.3 of Appendix B to Chapter 1 states that "the ALWR design will satisfy NRC requirements concerning automation of safety-related operator

actions and operator response times. Since that part of the issue concerning automatic realignment of ECCS is not applicable to the ALWR by design, this issue is considered resolved for the ALWR."

Section 3.2.2 of Chapter 5 states that "the source of core coolant inventory makeup shall be adequate for at least 36 hours without need for manually switching suction sources for non-LOCA events."

Staff Evaluation: In the DSER for Chapter 10, the staff concluded that EPRI had not provided the technical basis or specific requirements that support its assertion that the quantitative standards and the process requirements in the Evolutionary Requirements Document are significantly more limiting than those in American National Standards Institute/American Nuclear Society (ANSI/ANS) 58.8-1984, "Time Response Design Criteria for Safety Related Operator Actions at Nuclear Power Plants." In response to the DSER, EPRI has modified the Requirements Document to provide the necessary technical basis and specific requirements. In addition, EPRI has committed to satisfy NRC requirements concerning automation of safety-related operator actions and operator response times. Since the revision to ANSI/ANS-58.8 (now expected to be issued in April 1993) is not expected to alter the time response criteria of the 1984 edition of the standard, the staff concludes that EPRI has presented an appropriate resolution of this issue for ALWR evolutionary plant designs.

Issue B-17 has not yet been generically resolved by the staff and has a medium safety priority. Should the final agency resolution of this issue dictate additional actions for ALWR evolutionary plant designs, the staff will expect the FDA/DC applicant to address resolution of Issue B-17 for staff evaluation.

3.2.12 B-32, Ice Effects on Safety-Related Water Supplies

Issue: The service water system (SWS) rejects heat to the ultimate heat sink, which, during an accident or transient, cools the component cooling water heat exchangers. This in turn cools the residual heat removal heat exchangers, as well as provides cooling for safety-related pumps and area cooling coils. Blockage of the SWS flow intake structure by ice has led to plant shutdowns, reduced power operation for repairs and modifications, and degraded modes of operation.

EPRI Proposal: EPRI states that the problems associated with unreliable water flow from ultimate heat sinks as a result of fouling, blockage, and pump failure are addressed by the design requirements and guidance in the Evolutionary Requirements Document. The plant designer is given specific direction throughout Chapter 8 of the Evolutionary Requirements Document to design the ultimate heat sink and related water systems for both safety-related and non-safety-related functions to minimize or avoid these problems. The requirements in Chapter 8 that are intended to resolve the issue of ice blockage of the SWS flow intake include the following:

- Section 7.3.1.3 requires that the safety heat sink be capable of performing its intended function during freezing weather. System features will be provided to prevent the pond from freezing. The plant designer will provide the utility with supporting analyses showing that the safety heat

sink will not freeze during the winter. Weather conditions from the site survey will be used for this analysis, and the effects of icing of the intake structure will be considered.

- Section 7.3.2.2 requires the recirculation of warm water through the intake structures for anti-icing protection.

The Evolutionary Requirements Document also directs the plant designer to further investigate the problem of ice as a flow-blockage mechanism and to provide provisions for disposing of and/or dissolving such ice, as required, and to design the ultimate heat sinks and water flow systems to avoid or minimize, as achievable, this problem.

Staff Evaluation: This issue has been subsumed by Issue 153. See Section 3.2.63.

3.2.13 B-55, Improved Reliability of Target Rock Safety/Relief Valves

Issue: The majority of valves in BWR pressure relief systems are Target Rock safety/relief valves, and a significant number of failures of these valves have occurred. Failures include valves (1) failing to open properly on demand, (2) opening spuriously and then failing to reseat properly, and (3) opening properly and then failing to reseat properly.

In the late 1970s, the NRC staff concluded that the inadvertent blowdown events that had occurred as a result of malfunctions of pressure relief system valves had neither significantly affected the structural integrity or capability of the reactor vessel or its internals or the pressure-suppression containment system, nor resulted in any significant radiation releases to the environment. Even if such events were to occur more frequently, there would not likely be any significant effects. The performance of these valves, however, is under continual surveillance and the consequences of their failures are subject to review.

EPRI Proposal: In the note to Table B.3-1 in Appendix B to Chapter 1 of the Evolutionary Requirements Document, EPRI states that Issue B-55 was determined, through the NUREG-1197 screening process, to be not technically relevant to the ALWR design.

Staff Evaluation: Appendix A to NUREG-1197 identifies issues that the staff determined were not applicable to the ALWR Program according to six categories. Issue B-55 is listed as a Code (or Category) 3 item, that is, an issue that is applicable only to reactor types and design features not included in current standardized designs. It is the staff's understanding that Target Rock safety/relief valves are not to be used in any ALWR evolutionary plant designs.

Issue B-55 has not yet been generically resolved by the staff and has a medium safety priority. Should the final agency resolution of this issue dictate that it is otherwise applicable to ALWR evolutionary plant designs or indicate additional actions, or should Target Rock safety/relief valves be used in any evolutionary plant designs, the staff will expect the FDA/DC applicant to address the resolution of Issue B-55 for staff evaluation.

3.2.14 B-56, Diesel Reliability

Issue: If a loss of normally available (ac) power occurs at a nuclear power plant, redundant onsite emergency ac power sources provide power for necessary safety functions, which include reactor core decay heat removal (General Design Criterion (GDC) 34 of Appendix A to 10 CFR Part 50), emergency core cooling (GDC 35), and containment heat removal (GDC 38). The systems performing these functions are essential for preserving the integrity of the reactor core, reactor coolant system, and containment. Although reactor core decay heat can be removed for a limited time by systems that are independent of ac power, most licensees have selected Class 1E emergency diesel generators (EDGs) as the long-term ac power sources in most plants for meeting the requirements of 10 CFR 50.63. Therefore, the reliability of EDGs is a major factor in ensuring acceptable plant safety.

EPRI Proposal: Section 3.11.2 of Appendix B to Chapter 1 of the Evolutionary Requirements Document states:

- The NRC is currently in the process of upgrading Regulatory Guide 1.9, *Selection, Design, and Qualification of Diesel-Generator Units Used as Standby (Onsite) Electric Power Systems and [sic] Nuclear Power Plants*, to incorporate improvements in diesel reliability programs. Regulatory Guide 1.108 will be withdrawn, and revisions to SRP sections and Technical Specifications are being prepared. NUMARC [Nuclear Management and Resources Council] plans to revise Appendix D of NUMARC-87-00 to assure consistency with the emergency diesel generator reliability program described in Regulatory Guide 1.9.

- The ALWR is committed to improving the reliability of their EDG units over current experience and have required a minimum target reliability of 0.98. This will be achieved by only using EDG units, including auxiliary and support and control systems, that have demonstrated this reliability or, for new designs, are qualified per the latest revisions of ANSI/IEEE Standard 387 and IEEE [Institute of Electrical and Electronics Engineers] Standard 323. Additionally, the reliability of the EDGs will be improved by a design that allows longer start times, thus eliminating problems which adversely affect reliability. Finally, the EDGs will be backed up by combustion turbines which must also have a target reliability of 0.98. These requirements should resolve this issue for the ALWR Requirements Document.

Staff Evaluation: EPRI identifies a minimum EDG target reliability of 0.98 and a backup from combustion turbines having a target reliability of 0.98. However, targets alone do not ensure maintenance of acceptable reliability levels.

In the course of resolving Issue B-56, the Commission recently directed the staff to amend 10 CFR 50.63 to require monthly EDG surveillance testing and reporting of failures to start and load runs against criteria developed to identify potential degradation of EDG performance. This performance-based rule (when issued) would apply to LWRs. Comparable performance monitoring criteria would apply to ALWRs.

Since the Evolutionary Requirements Document does not provide sufficient information to enable the staff to fully evaluate the design requirements related to this generic issue, the staff will evaluate the design features for acceptability when it reviews individual applications for FDA/DC.

Issue B-56 has not yet been generically resolved by the staff and has a high safety priority. Should the final resolution of this issue dictate additional actions for ALWR evolutionary plant designs, the staff will expect the FDA/DC applicant to address the resolution of Issue B-56 for staff evaluation.

3.2.15 B-61, Allowable Emergency Core Cooling System Equipment Outage Periods

Issue: This issue entails the establishment of surveillance test intervals and allowable equipment outage periods for operating nuclear power plants, using analytically based criteria and methods for the technical specifications. The allowable outage and testing intervals in present technical specifications were determined primarily on the basis of engineering judgment.

EPRI Proposal: In the note to Table B.3-1 in Appendix B to Chapter 1 of the Evolutionary Requirements Document, EPRI states that Issue B-61 was determined, through the NUREG-1197 screening process, to be not technically relevant to the ALWR design.

Staff Evaluation: Appendix A to NUREG-1197 identifies issues that the staff determined were not applicable to the ALWR Program according to six categories. Issue B-61 is listed as a Code (or Category) 5 item, that is, an issue not applicable to plant design, such as plant operation and operating procedures, management of operations, accident management and emergency preparedness, operator training and qualifications, inspection and maintenance, and operating experience reporting. Since Issue B-61 does not appear to address plant design, the staff does not consider it to be applicable according to the definitions of Appendix A to NUREG-1197.

Issue B-61 has not yet been generically resolved by the staff and has a medium safety priority. Should the final resolution of this issue dictate that it is otherwise applicable to ALWR evolutionary plant designs or indicate additional actions, the staff will expect the FDA/DC applicant to address the resolution of Issue B-61 for staff evaluation.

3.2.16 B-64, Decommissioning of Reactors

Issue: This issue entails the need for rulemaking to provide regulations for the orderly retirement of nuclear facilities from service and the safe disposition of the remaining radioactivity.

EPRI Proposal: In the note to Table B.3-1 in Appendix B to Chapter 1 of the Evolutionary Requirements Document, EPRI states that Issue B-64 was determined, through the NUREG-1197 screening process, to be not technically relevant to the ALWR design.

Staff Evaluation: According to Appendix B to NUREG-0933, Issue B-64 is not applicable to future plants; therefore, an evaluation is not necessary.

3.2.17 C-8, Main Steamline Isolation Valve Leakage Control Systems

Issue: This issue was initiated to investigate the desirability of the main steam isolation valve leakage control system (MSIVLCS) recommended in RG 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants," for BWRs. After it was identified, the staff determined that the issue was of little or no significance to plant risk. New concerns arose from subsequent operational experience that indicated a relatively high MSIVLCS failure rate and a variety of failure modes at some BWR plants. Additional data on the magnitude and frequency of MSIV leakage at BWRs renewed concerns about the viability of the MSIVLCS design. In addition, the question of backfitting MSIVLCSs for BWRs was raised.

EPRI Proposal: Section 3.3.2 of Appendix B to Chapter 1 of the Evolutionary Requirements Document states that Issue C-8 is addressed in Section 2.3.1 of Appendix B as an ALWR optimization subject. EPRI proposes to eliminate the safety-related leakage control system to allow higher leakage rates through the MSIV and to use an alternative MSIV leakage treatment method that takes credit for the large surface volume in the main steam piping and the condenser hotwell to plate out the fission products following core damage.

Staff Evaluation: As discussed in Section 3.1 of this appendix, Issue C-8 is not included in Appendix B to NUREG-0933 because the staff had determined that is was not applicable to future plants. In its DSER for Chapter 3, however, the staff evaluated Issue C-8 and indicated that it did not expect to reach a conclusion regarding EPRI's proposal until the issue is generically resolved. The staff's review of the MSIVLCS is given in Section 2.3.1 of this appendix. On the basis of that review, this issue is resolved.

3.2.18 2, Failure of Protective Devices on Essential Equipment

Issue: Essential equipment has experienced a large number of failures or has been incapacitated as a result of either the failure or intentional bypass of protective devices intended to trip active engineered safety features (ESFs) for indications of equipment faults. The affected systems exist throughout the plant and include the plant control system, the plant protection system, and the ESFs. The staff is concerned that the reliability estimates for essential equipment may not properly account for failure of the protective devices. Because of the loss of redundant devices through failures of circuits intended to be independent, there is an increased probability of common-mode failure of redundant vital services. This issue needs to be studied further to determine if failure rate estimates for essential equipment have increased and if essential equipment could be made significantly more reliable by improving the reliability of protective devices.

EPRI Proposal: The Evolutionary Requirements Document includes a number of specific design requirements to address the safety concerns of this issue. Chapter 5 requires that

- plant systems embody sufficient robustness of design to tolerate a conservative number of spurious or inadvertent engineered safety system actuations without the need for followup tests or inspections to verify the systems' integrity or operability (Section 2.2.7)

- each division of engineered safety systems be totally independent electrically unless it is otherwise physically impracticable or less safe (Section 2.3.2)

Chapter 10 requires that

- a robust system design, including segmentation of major functions, separation of redundant equipment within a segment, and fault-tolerant equipment, be developed to achieve high reliability and protection against the propagation of faults (Section 2.2.3)

- the man-machine interface system (M-MIS) equipment be designed and configured to readily support inservice testing by providing built-in test features, incorporating good human factors principles, and avoiding the use of undesirable features such as test jumpers or lifting of leads (Section 2.2.4)

- the M-MIS design process explicitly consider the potential for and the consequences of failures of plant and M-MIS components (Section 3.1.3.4)

- the capability for continuous on-line self-testing of hardware integrity be provided for as much of the M-MIS as is practicable (Section 3.6.1)

- on detection of a failure in the M-MIS, a system be designed so that it can be placed in such configuration that an additional single failure will not prevent system-level protection or safety action (Section 3.6.4)

- the mean time to detect and repair failures down to the lowest replaceable module, averaged across all types of M-MIS equipment for the entire design life, be less than 4 hours and the maximum time for any module be less than 8 hours (Section 3.7.4)

Staff Evaluation: Section 2.3.2 of Chapter 5 of the Evolutionary Requirements Document states that each division of the ESF systems needs to be electrically independent unless impracticable or less safe. In its DSER for Chapter 10, the staff agreed that the implementation of the design criteria in the Evolutionary Requirements Document would resolve this issue, with the exception of the staff's concern about lack of specific guidelines for exceptions to the independence criteria for electrical systems. Since it has not prioritized Issue 2, the staff will defer further evaluation of the open issue in the DSER to its review of individual applications for FDA/DC.

3.2.19 15, Radiation Effects on Reactor Vessel Supports

Issue: Neutron damage of reactor vessel structural materials causes embrittlement that may increase the potential for propagation of flaws that may exist in the materials. As long as the operating environment in which the materials are used has a higher temperature than the materials' nil ductility transition temperature (i.e., the lowest temperature at which the materials would not be susceptible to failure by brittle fracture), no failure by brittle fracture is expected. When subjected to neutron irradiation, many materials become more susceptible to brittle fracture at the operating temperatures of interest, which should have been accounted for in the design and fabrication of the reactor vessel support structures.

Research has indicated that the susceptibility of materials exposed to neutron irradiation has been underestimated. Loss of material fracture toughness may result in failure of the reactor vessel support structures and consequent reactor vessel movement as a result of transient stress or shock, such as would be experienced in a seismic event. Reactor vessel support structures in nuclear power plants using particular types of supports (long-column, shield-tank, short-column, suspension) are assumed to be susceptible to radiation damage.

EPRI Proposal: Proposed elements for resolving this issue are included in Section 3.4.1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document.

Staff Evaluation: The elements of resolution proposed in the Evolutionary Requirements Document appear to adequately address the safety concerns of this issue. However, Issue 15 has not yet been generically resolved by the staff and has a high safety priority. Since it has not yet finalized its position on this issue for ALWRs, the staff will evaluate the applicant's proposed resolution of Issue 15 during its review of individual applications for FDA/DC.

3.2.20 23, Reactor Coolant Pump Seal Failures

Issue: This issue addresses the high rate of reactor coolant pump (RCP) seal failures that challenge the makeup capacity of the emergency core cooling systems in PWRs.

EPRI Proposal: Section 3.3.1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document provides the elements for resolving this issue. EPRI revised Chapter 3 of the Evolutionary Requirements Document to require that degradation of the shaft seal system be negligible following loss of both seal injection and pump cooling water for up to 1 hour. Also, in the event of loss of all ac power, reactor coolant system leakage through the RCP shaft will be limited so that the reactor core remains covered and natural circulation is maintained for at least 8 hours.

Staff Evaluation: In its DSER for Chapter 3, the staff concluded that some of the design requirements proposed by EPRI might not be practical unless adequate pump seals were developed and/or independent seal cooling was provided. In particular, the staff pointed out that Section 3.4.2.2.5 of Chapter 3 of the Evolutionary Requirements Document (at the time the DSER was prepared) stated that during station blackout conditions, the system design leakage through the pump seals is to be limited to 8 gpm per pump to ensure that the reactor core remains covered for 8 hours. The staff believes that this requirement would be difficult to meet, given the present hydrostatic RCP seal. This seal requires an injection cooling supply of approximately 8 gpm and, if uncooled, the seal leakage flow could increase to 21 gpm. Hydrodynamic-type seals might meet the 8-gpm limit with the leakage uncooled; however, this could probably not be ensured for an extended station blackout. These leakage figures are for an intact seal system. Without seal cooling, the seal may fail ("pop open"), which could lead to leakage rates of several hundred gallons per minute.

Issue 23 has not yet been generically resolved by the staff and has a high safety priority. Unless the final agency resolution of this issue dictates other actions for ALWR evolutionary plant designs, the staff will expect the FDA/DC applicant to submit a design that provides for independent RCP seal cooling during station blackout or to provide adequate testing of the proposed seal design to demonstrate integrity following extended loss of seal injection and cooling.

3.2.21 24, Automatic Emergency Core Cooling System Switch to Recirculation

Issue: The emergency core cooling system (ECCS) has two different phases, the injection phase and the recirculation phase. The injection phase involves initial cooling of the reactor core and replenishment of the primary coolant following a loss-of-coolant accident. The recirculation phase provides long-term cooling during the accident recovery period.

Switchover from the injection phase to the recirculation phase includes alignment of a number of valves to the recirculation position. Switchover can be achieved by a number of manual actions (manual option), by automating these actions (automatic option), or by automatic realignment of certain valves and manual completion of the switchover process (semiautomatic option). These three switchover options are vulnerable in varying degrees to human errors and hardware failures, as well as common cause failures. Although an automatic system designed to control the whole switchover process or a portion of it can reduce the impact of operator error in executing the switchover, automatic systems may be subject to spurious actuation. Spurious switchover of ECCS and containment spray pump suction to a dry containment sump can result in pump damage and possible loss of safety function, resulting in potentially unacceptable safety consequences. Review of past reactor experience indicates the existence of a significant number of ECCS spurious actuations, such as the four that occurred in 1980 at the Davis-Besse Nuclear Plant, Unit 1.

EPRI Proposal: Not provided in Section 3 of Appendix B to Chapter 1 of the Evolutionary Requirements Document.

Staff Evaluation: Section 3.1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document states that it specifically addresses compliance with high- and medium-priority and nearly resolved generic safety issues included as of January 1990 in the NRC's generic issue management control system (GIMCS). As of December 31, 1989, although Issue 24 was identified in Appendix B to NUREG-0933 as being applicable to future plants, it had not yet been prioritized.

As of the first-quarter fiscal year 1992 update of the GIMCS report, Issue 24 has a medium safety priority. Since this issue is not addressed in the Evolutionary Requirements Document, the staff will evaluate its proposed resolution during its review of individual applications for FDA/DC.

3.2.22 29, Bolting Degradation or Failure in Nuclear Power Plants

Issue: Issue 29 was established in 1982 to address staff concerns about the number of events involving the degradation of threaded fasteners (bolt cracking, corrosion, failure, etc.) in operating plants from 1964 to the early 1980s. Many of the events were related to components of the reactor coolant pressure boundary (RCPB) and support structures of major components. This

raised questions regarding the integrity of the RCPB and the reliability of the component support structures following a loss-of-coolant accident or a seismic event.

Failures reported by licensees involved a variety of threaded fasteners and several causes. As a result, several different failure mechanisms had to be considered. Most frequent were wastage (corrosion) from boric acid attack and stress corrosion cracking (SCC). The former occurred more often at RCPB joints; the latter in structural bolting.

EPRI Proposal: Proposed elements for resolving this issue are included in Section 3.1.1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document.

Section 3.1.1.3 of Appendix B to Chapter 1 provides a discussion and regulatory status of Issue 29 and summarizes data on industry experience with bolting degradation compiled by the NRC (NUREG-0943 and NUREG-1095) and the industry (Institute of Nuclear Power Operations SOER 84-5, "Bolting Degradation or Failure in Nuclear Power Plants," and EPRI NP-3784, "A Survey of the Literature on Low-Alloy Steel Fastener Corrosion in PWR Power Plants"). Also mentioned is the cooperative industry effort of the Atomic Industrial Forum/Metal Properties Council Task Group on Bolting and EPRI to resolve the bolting issue (which resulted in EPRI NP-5769, "Degradation and Failure of Bolting in Nuclear Power Plants"). Results of EPRI NP-5769 and its NRC evaluation (NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants") are summarized. NRC's intent to revise the Standard Review Plan to provide criteria for evaluating the bolting practices proposed by applicants for new plants is also mentioned.

Section 3.1.1.4.2 of Appendix B to Chapter 1 provides requirements for all ALWRs regarding Issue 29. These requirements in Chapter 1 of the Evolutionary Requirements Document are the following:

- Section 2.2.3 requires that the use of best-available materials and water chemistry be specified, based on the extensive LWR operating experience.

- Section 5.2.1.1 requires that the materials used be selected from materials proven in service in LWR plants. Proven materials are those with the same nominal composition and subsequent processing steps and same exposure conditions as those used successfully for at least several years in existing LWRs.

- Section 5.2.4.3 requires that ALWR designs avoid very high strength fastener materials and make use of conventional materials applied well within the limits and for which successful experience has been obtained. In particular, the use of high-strength bolts or fasteners will be eliminated, where practicable, and sufficiently robust designs will be used so that special, high-strength materials are not required.

- Section 5.3.3.1 requires that the materials for threaded fasteners used to maintain pressure boundary integrity in the reactor coolant and related systems and in the steam, feed, and condensate systems; threaded fasteners used inside those systems; and threaded fasteners used in pipe and component structural mountings for those systems be selected and specified on the basis of their previous satisfactory performance in

similar applications. Similarity will be based on comparable temperature and environment; comparable stresses; comparable service cycles; comparable fabrication and installation; and comparable inspection during fabrication and installation and while in service. The application of threaded fasteners will be in accordance with the requirements of Section 12.7 of Chapter 1 and with the guidelines of EPRI NP-6316, "Guidelines for Threaded-Fastener Application in Nuclear Power Plants."

- Section 5.3.3.2 requires that the lubricants used on all threaded fasteners that maintain pressure boundary integrity in the reactor coolant and related systems and in the steam, feed, and condensate systems; threaded fasteners used inside those systems; and threaded fasteners used in pipe and component structural supports for those systems be completely specified by the plant designer in appropriate drawings and specifications; that is, field selection of thread lubricants is not permitted. The thread lubricants will be selected on the basis of experience and test data that show they are effective, but do not cause or accelerate corrosion of the fastener. Where leak sealants are used on threaded fasteners or can contact the fastener in service, their selection will be based on satisfactory experience or test data. Possible adverse interaction between sealants and lubricants should be considered.

- Section 5.5.1 requires that materials selected for use in the ALWR be compatible with the full range of environmental conditions that may be encountered over the plant life. These environmental conditions include chemical (e.g., fluids that contact the material), temperature, and radiation factors. The plant designer will document the environmental conditions used as the basis for the selection of ALWR materials. These environmental conditions will be consistent with the specific requirements in other chapters of the Evolutionary Requirements Document.

- Section 12.7.1.1 requires that bolted joints and threaded fasteners be used only where a clear need for them has been established. The use of a bolted joint should include consideration of the initial cost, the subsequent maintenance effort, and the risk of service problems that such joints introduce. Where a bolted joint must be used to satisfy functional requirements, its adequacy should be achieved preferably by a simple, rugged design that does not depend on high-strength or unusual materials, precision parts, or specialized assembly and inspection methods.

- Section 12.7.1.4 requires that the configuration of bolted joints be based on their previous satisfactory performance in similar applications and the current practice at the time the plant is designed. Similarity will be based on comparable materials; comparable temperature and local environment; comparable stresses; comparable loading and service cycles; comparable fabrication and installation; comparable inspection during fabrication and installation and while in service; and comparable methods of locking or retention of threaded fasteners.

- Section 12.7.2.1 requires that where the bolted joint provides sealing of a pressure boundary (and especially where the contained material is radioactive, toxic, or corrosive), the joint configuration be designed to

minimize the potential for leakage, to minimize the consequences of such leakage (e.g., by using corrosion-resistant materials), and to facilitate the detection of any leakage.

- Section 12.7.2.1.1 requires that in those applications where gasket leakage can lead to corrosion and significant structural degradation of the joint (e.g., borated water leakage on carbon steel), joint parts with adequate corrosion resistance be used. However, functional adequacy of the joint must not be compromised; that is, the most important attribute is that the joint be leaktight.

- Section 12.7.2.1.2 requires that if corrosion-resistant fasteners cannot be used in a particular joint and leakage could result in corrosion, inspection and maintenance capability be provided. Any such joints must be identified by the plant designer to the utility, in writing, along with the basis for their acceptability and how inspection and maintenance would be accomplished.

- Section 12.7.2.2 requires that the joint configuration, including threaded fasteners, facilitate inspection of the condition of the joint; for example, direct visual inspection of the condition of the load-carrying portions of the joint should be practical without disassembly.

- Section 12.7.2.3 requires that the joint design specifically allow for tolerances in fabrication and variability in installed preload, as well as changes in preload that may occur in service (e.g., relaxation).

- Section 12.7.2.4 requires that where a bolted joint uses preloaded fasteners, the plant designer specify, in appropriate documents (drawings or procedures), the preload required and how it is to be achieved, including, for example, thread lubricants and torquing sequences. The setting of preloads and the selection of methods of preloading should not be left to the choice of shop or field activities.

- Section 12.7.3.1 requires that threaded fasteners be locked under the following conditions:

  - where the threaded fasteners are subjected to alternating loads that are a significant fraction of their minimum preload and especially where the fastener's preload could be released by small deformations of the fastener or joint parts

  - where the threaded fastener cannot be preloaded significantly even though the expected loads are small

- Section 3.1.1.4.3 states that the Evolutionary Requirements Document addresses all elements considered for the resolution of Issue 29 and that Issue 29 is resolved for the ALWR.

<u>Staff Evaluation</u>: The staff technically resolved Issue 29 in October 1991 with the issuance of GL 91-17, "Generic Safety Issue 29, 'Bolting Degradation or Failure in Nuclear Power Plants,'" and NUREG-1339. The resolution is based on (1) operating experience with bolting in both nuclear and conventional power plants; (2) actions already taken through bulletins, generic letters,

and information notices since 1982; and (3) industry-proposed recommendations and actions, which are documented in EPRI NP-5769 and NP-5067, "Good Bolting Practices, A Reference Manual for Nuclear Power Plant Maintenance Personnel."

Even though the EPRI proposal discussed above appears to have met most of the recommendations in GL 91-17, the staff is not in total agreement with the proposal, as discussed below.

- Section 3.1.1.3 should reference GL 91-17 and discuss compliance with GL 91-17 as the basis for the technical resolution of Issue 29.

- The leaktightness of a bolted joint that provides sealing of a pressure boundary is significantly affected by whether the joint is properly assembled or not. If a bolted joint is to be successful, preloads developed at assembly must be sufficient to maintain joint integrity throughout the service life under external loads and service conditions. EPRI NP-5067, Volumes 1 and 2, provides, among other things, good bolting practices for assembling bolted joints and should be referenced in Section 3.1.1.3, as was done in GL 91-17.

- When Issue 29 was being resolved, the staff addressed several specific issues on threaded fasteners in bulletins, generic letters, and information notices (e.g., PWR coolant pressure boundary bolting and component degradation due to boric acid corrosion; stress corrosion cracking of internal bolting in certain types of check valves; traceability and material control of fasteners; and nonconforming, misrepresented, counterfeit, and/or fraudulent bolting), the details of which can be found in NUREG-1339. The pertinent information and requirements in these NRC bulletins, generic letters, and information notices issued to operating plant owners regarding Issue 29 should be factored into the discussion of the issue in the Evolutionary Requirements Document.

Since the Evolutionary Requirements Document does not provide sufficient commitments or information regarding these concerns and since it does not provide sufficient design details about the selection of bolting materials, lubricants, and leak sealants, the staff will evaluate resolution of these concerns during its review of individual applications for FDA/DC.

3.2.23 51, Improving the Reliability of Open-Cycle Service Water Systems

Issue: Issue 51 was established because fouling of safety-related service water systems (SWSs) by either mud, silt, corrosion products, or aquatic bivalves had led to plant shutdowns, reduced power operation for repairs and modifications, and degraded modes of operation in nuclear power plants.

EPRI Proposal: The only generic safety issue related to SWS reliability addressed in Appendix B to Chapter 1 of the Evolutionary Requirements Document is Issue 130, "Essential Service Water Pump Failures at Multiplant Sites" (see Section 3.8.1 in Appendix B to Chapter 1 of the Evolutionary Requirements Document and Section 3.2.57 of this appendix). However, Table B.1-2 in Appendix B to Chapter 1 indicates a commitment to comply with GL 89-13, "Service Water Systems Problems Affecting Safety Related Equipment," the closure document for the technical resolution of Issue 51 for existing plants. EPRI considers GL 89-13 to be appropriate for ALWRs.

The requirements in Chapter 8 of the Evolutionary Requirements Document that are intended to resolve this issue include the following:

- Section 1.3.3 states that the SWS will be an open system with its functions classified as both safety related and non-safety related.

- Section 1.3.5 states that the normal power heat sink will be the source for plant circulating water and non-safety service water. The ultimate heat sink (UHS) will be the source for safety service water and will act as the repository for heat rejected from safety-related and non-safety-related reactor auxiliary plant systems and components during all modes of plant operation. The UHS is assumed to be a cooling pond.

- Section 1.5.1 states that a goal of the designs referencing the Evolutionary Requirements Document will be to minimize the possible effects on the plant cooling water systems of mud, silt, organisms, and harsh chemistry of heat sink water.

- Section 1.5.4 requires that direct service water not be used for component cooling. A closed-loop component cooling water system (CCWS) will be used to transfer heat from the component heat loads via a heat exchanger to the SWS and the heat sink.

- Section 2.2.6 requires that the cooling water piping from the heat sink to the SWS be equipped with a filtration system. Provisions for periodic cleaning and backflushing, as necessary for the heat transfer and piping surfaces of the SWS, will also be included. Where required, chemical injection will be included for the removal of organisms.

- Section 2.2.7 requires that the systems' designs provide for periodic testing to ensure system adequacy and complete post-maintenance verification of operability.

- Section 2.2.11 requires that all systems be inspectable and testable for operability and cooling capability by using normal and/or periodic testing.

- Section 3.3.2.1 specifies appropriate margin in the heat exchanger area to account for fouling and tube plugging. This margin permits a degradation in performance before tube cleaning or retubing is required.

- Section 3.3.2.2 specifies high- and low-flow velocities in heat exchanger tubes to minimize the potential for erosion or microbiologically induced corrosion.

- Section 3.3.2.4 requires that in the shell and tube SWS heat exchanger design, the SWS flow be routed through the tube side to facilitate the control and removal of fouling.

- Section 3.5.1 requires performance tests of key heat exchangers during preoperational testing if heat load is available. The plant designer is required to identify all heat exchangers that cannot be performance tested during preoperation tests and to develop a test plan and schedule for testing when heat load becomes available.

- Sections 3.6, 3.7.2, and 3.7.3 require the plant designer to select materials and water chemistry for cooling water systems to account for normal corrosion-erosion as well as those factors unique to cooling water systems, such as microbiologically induced corrosion, high purity water, or long layups. Additional requirements to address the issue of corrosion-erosion include the following:

  - Evaluate the addition of chemicals to the SWS to halt microbiological corrosion or to kill organisms in the water.

  - For the CCWS, provide for the addition of chemicals to the high purity water to reduce corrosion, and recommend the use of wet layup to reduce long-term corrosion of system heat exchangers.

  - Evaluate the types of service water generally available for plant cooling, and prepare a list of recommended system materials for piping, pumps, and heat exchangers operating under those conditions.

The Evolutionary Requirements Document states that eliminating the possible effects of fouling and blockage of the SWS and heat sinks is a design goal. The plant designer is given specific requirements and guidance for achieving this goal, including instructions to consider designs and new requirements that further mitigate the effects of fouling. The final design of the ALWR heat sinks and water flow systems will avoid or minimize, as achievable, the problems described in this section. EPRI's position is that implementation of these requirements should resolve Issue 51 for the ALWR.

Staff Evaluation: The staff's DSER for Chapter 8 contained two open issues related to biofouling. Although the Evolutionary Requirements Document includes a commitment to comply with GL 89-13, the concerns in the DSER, which have not yet been addressed by EPRI, are as follows:

- Enclosure 1 of GL 89-13 recommends that an ongoing program of surveillance and control (biocide, chemical) be implemented and maintained to reduce fouling problems in open-cycle SWSs. Section 5.4.3 of Chapter 8 of the Evolutionary Requirements Document discusses surveillance for reducing fouling, but not how to control it. Control of biofouling by the addition of biocides or chemicals should be included in this section, together with surveillance, as the two main elements for reducing fouling. The addition of biocides and chemicals to raw service water is discussed in Section 3.7 of Chapter 8 of the Evolutionary Requirements Document, but that section should be referenced in Section 5.4.3, and vice versa. Biocide injection into raw service water before long wet layups is mentioned in Section 3.7.2. It should also be mentioned in Section 5.4.3 as an effective means of preventing microbiologically influenced corrosion.

  Since the Evolutionary Requirements Document still does not provide sufficient commitments, information, or design details regarding this open issue, the staff will evaluate its resolution during its review of individual applications for FDA/DC.

- Since GL 89-13 was issued, the staff became aware of the potential problems for plants in the Great Lakes area related to biofouling of SWSs and CCWSs caused by a recently identified biofouling agent, the zebra

mussel. The staff issued NRC Information Notice (IN) 89-16, "Biofouling Agent: Zebra Mussel," to alert utilities to the tenacious nature and the unusually swift spread of those mussels in the Great Lakes area since they were first discovered there in 1988. Recently, a number of nuclear power plants in the Great Lakes area, including Fermi Unit 2 and D. C. Cook, have experienced biofouling problems due to zebra mussels in their cooling water systems and/or intake structures. The staff will review the potential for biofouling of SWSs and CCWSs for plants sited in the Great Lakes area on a case-by-case basis.

The DSER also contained an open issue related to heat exchanger testing. In the DSER, the staff concluded that the vendors should consider including the capability for performance testing of key heat exchangers during the planning and design stage. Section 3.5.1 of Chapter 8 of the Evolutionary Requirements Document contains provisions that address the staff's concerns. Therefore, this open issue is resolved.

In addition to the above, Table B.1-2 in Appendix B to Chapter 1 of the Evolutionary Requirements Document should be corrected to show that Chapter 8 is the lead chapter for GL 89-13 instead of Chapter 9. Also, "RESOLVED ISSUE (51)" should be added to the comments column.

3.2.24 57, Effects of Fire Protection System Actuation on Safety-Related Equipment

Issue: This issue addresses fire protection system (FPS) actuations that have resulted in adverse interactions with safety-related equipment at operating nuclear power plants. Events have shown that safety-related equipment subjected to FPS water spray could be rendered inoperable and that numerous spurious actuations of the FPS have been initiated by operator testing errors or by maintenance activities, steam, or high humidity in the vicinity of FPS detectors. The NRC issued Office of Inspection and Enforcement (IE) Information Notice 83-41, "Actuation of Fire Suppression System Causing Inoperability of Safety-Related Equipment," to alert licensees and provide recent examples in which FPS actuations caused damage or inoperability of systems important to safety. In addition, the staff is considering the need for modifying FPS requirements or licensing review procedures.

EPRI Proposal: Proposed elements for resolving this issue are included in Section 3.9.1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document.

Staff Evaluation: The elements of resolution proposed in the Evolutionary Requirements Document that reference Chapter 9 are evaluated by the staff in Section 3 of Chapter 9 of this report. These elements and the others in Section 3.9.1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document appear to adequately address the safety concerns of this issue. However, Issue 57 has not yet been generically resolved by the staff and has a medium safety priority. Since it has not yet finalized its position on this issue for ALWRs, the staff will evaluate the resolution of Issue 57 during its review of individual applications for FDA/DC.

3.2.25 70, Power-Operated Relief Valve and Block Valve Reliability

Issue: Power-operated relief valves (PORVs) and block valves were originally designed as non-safety-grade components in the reactor pressure control system for use only when plants are in operation; the block valves were installed because of expected leakage from the PORVs. Neither valve type was required to safely shut down a plant or mitigate the consequences of accidents. In 1983 the staff determined that PORVs were relied on to mitigate design-basis steam generator tube rupture accidents and questioned the acceptability of relying on non-safety-grade components to mitigate design-basis accidents. This issue addressed assessing the need for improving the reliability of PORVs and block valves.

EPRI Proposal: Proposed elements for resolving this issue are included in Section 3.5.4 of Appendix B to Chapter 1 of the Evolutionary Requirements Document. Section 5.5 of Chapter 5 establishes requirements for a safety-grade safety depressurization and vent system (SDVS) for ALWR PWRs.

Staff Evaluation: In its DSER for Chapter 5 (February 1990), the staff concluded that EPRI's proposed resolution for Issue 70 was consistent with the staff's positions. However, the staff concluded that the Evolutionary Requirements Document should specify a criterion for the depressurization rate of the SDVS. In Section 6.6.4.3 of Chapter 5, EPRI specifies that the SDVS will be capable of depressurizing the reactor coolant system below 250 psig before reactor vessel melt-through as a means of precluding containment challenges through direct containment heating. In Section 5.4 of Chapter 5 of this report, the staff concludes that this is an acceptable design objective. Therefore, this open issue is resolved.

The staff has resolved Issue 70 and has issued GL 90-16, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,'" and Generic Issue 94, "Additional Low Temperature Overpressure Protection for Light-Water Reactors" (requiring technical specifications revisions at PWRs with PORVs and block valves). It has also drafted revisions of the Standard Review Plan (Sections 3.2.2, "System Quality Group Classifications"; 5.2.2, "Overpressure Protection"; and 5.4.7, "Residual Heat Removal System").

The low-temperature overpressure protection (LTOP) function for the ALWR PWRs is to be provided by the residual heat removal pressure-relief system. In a letter to the NRC dated March 28, 1988, EPRI committed to revise appropriate chapters of the Evolutionary Requirements Document to provide information on LTOP. The staff has confirmed the incorporation of this information in Section 3.3.2 of Chapter 3 of the Evolutionary Requirements Document, and the staff evaluation is documented in Section 3.3 of Chapter 3 of this report. Therefore, this confirmatory issue is resolved.

3.2.26 73, Detached Thermal Sleeves

Issue: Fatigue failure problems associated with nozzle-thermal sleeve assemblies have been identified in piping systems of both BWRs and PWRs.

EPRI Proposal: None provided in Section 3 of Appendix B to Chapter 1 of the Evolutionary Requirements Document.

Staff Evaluation: Section 3.1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document states that it specifically addresses compliance with the high- and medium-priority and nearly resolved generic safety issues included as of January 1990 in NRC's generic issue management control system (GIMCS). As of December 31, 1989, although Issue 73 was identified in Appendix B to NUREG-0933 as being applicable to future plants, it had not yet been prioritized.

As of the first-quarter fiscal year 1992 update of the GIMCS report, Issue 73 has a safety priority of nearly resolved. Consequently, the staff will evaluate the resolution of Issue 73 during its review of individual applications for FDA/DC.

3.2.27 75, Generic Implications of ATWS (Anticipated Transient Without Scram) Events at the Salem Nuclear Plant

Issue: On two occasions in 1983, Salem Unit 1 failed to scram automatically because both reactor trip breakers failed to open on receipt of an actuation signal. In both cases, the unit was successfully tripped by manual action. The failure of the breakers was attributed to excessive wear from improper maintenance of the undervoltage relays that receive the trip signal from the protection system and result in the breakers opening mechanically. Three separate NRC actions were initiated to address this problem. One was plant specific and was addressed before the restart of Salem Unit 1. The second action was an investigation of the Salem events and the circumstances leading to them.

The third action was the formation of an NRC task force to study the overall generic implications of this event. The results of the task force's work were reported in NUREG-1000, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant." This action resulted in the issue being broken down into 16 parts. Four parts of the issue required actions by licensees and applicants and were resolved with the issuance of GL 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events," and IE Bulletins 83-01, "Failure of Reactor Trip Breakers (Westinghouse DB-50) To Open on Automatic Trip Signal," and 83-04, "Failure of Undervoltage Trip Function of Reactor Trip Breakers." The remaining 12 parts of the issue were staff actions that have been resolved through licensing actions (7 parts), coverage in the Human Factors Program Plan (2 parts), issuance of IE Bulletin 83-08, "Electrical Circuit Breakers With an Undervoltage Trip Feature in Use in Safety-Related Applications Other Than the Reactor Trip System" (1 part), resolution of an unresolved safety issue (1 part), and staff plans to revise RG 1.33, "Quality Assurance Program Requirements Operation" (1 part).

EPRI Proposal: Proposed elements for resolving this issue are included in Section 3.10.1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document.

Staff Evaluation: The staff has resolved Issue 75 and has issued GL 83-28 and IE Bulletins 83-01 and 83-04. Concerns regarding diversity issues resulting from the ATWS rule are discussed in the DSER for Chapter 5 (February 1990), SECY-90-016 (January 12, 1990), and the Commission's staff requirements memorandum concerning SECY-90-016 (June 26, 1990) and are not addressed here.

In the DSER for Chapter 10, the staff reported on its review of EPRI's proposals addressing this issue against the requirements for existing plants in GL 83-28.

With regard to GL 83-28, Item 1.1 ("Post-Trip Review - Program Description and Procedure"), the staff stated that it would address this matter during its review of individual applications for FDA/DC, since EPRI stated that the program and procedures used by the plant owner to assess unscheduled reactor shutdowns were not within the scope of the Evolutionary Requirements Document.

Regarding GL 83-28, Item 2.1 ("Equipment Classification and Vendor Interface - Reactor Trip System Components") and Item 2.2 ("Equipment Classification and Vendor Interface - Programs for All Safety-Related Components"), the staff concluded in the DSER that the design criteria in the Evolutionary Requirements Document were acceptable, but by themselves, did not provide sufficient information for the staff to complete its review of this issue. The staff stated that it would address these matters during its review of individual applications for FDA/DC.

Regarding GL 83-28, Item 3.1 ("Post-Maintenance Testing - Reactor Trip System Components") and Item 3.2 ("Post-Maintenance Testing - All Other Safety-Related Components"), the staff stated in the DSER that the EPRI requirements for post-installation validation tests and long-term surveillance and maintenance were acceptable for the definition-of-design phase. However, the staff asked EPRI to perform a quantitative analysis of plant risk to justify its 14-day design objective for corrective maintenance of the man-machine interface system equipment discussed in Section 3.5.2.3 of Chapter 10 of the Evolutionary Requirements Document, including the protection system, the plant control system, and the plant information and monitoring systems. EPRI stated (letter to the NRC dated January 28, 1992) that the 14-day requirement in Section 3.5.2.3 was based on the premise that the maintenance technicians will perform a corrective maintenance operation every 2 weeks on the entire population of these systems. Such a maintenance requirement would not be an unreasonable burden on the plant staff and did not subvert the stringent requirements on system reliability. In addition, EPRI changed the first part of the first sentence of Section 3.5.2.3 of Chapter 10 to read: "Corrective maintenance of any of the following major parts...." This change is acceptable because it does not conflict with current regulatory guidance. Therefore, this open issue is closed. However, the staff will evaluate this item in detail during its review of individual applications for FDA/DC.

Regarding GL 83-28, Item 4.2 ("Reactor Trip System Reliability - Preventative Maintenance and Surveillance Program for Reactor Trip Breakers") and Item 4.5 ("Reactor Trip System Reliability - System Functional Testing"), the staff concluded in the DSER that EPRI's design criteria were acceptable (except as noted in Sections 3.6 and 8.3.4 of Chapter 10 of the DSER), but by themselves, did not provide sufficient information for the staff to complete its review of these items. The staff stated that it would address these items during its review of individual applications for FDA/DC.

3.2.28 76, Instrumentation and Control Power Interactions

<u>Issue</u>: Concerns have been raised regarding the potential for accidents or transients being made more severe as a result of control system failures, including control and instrumentation power supply faults. During the

licensing review process, the NRC staff performs an audit review of the non-safety-grade control systems to ensure that an adequate degree of separation and independence is provided between these non-safety-grade systems and the safety systems. On this basis, it is generally believed that control system failures are not likely to result in loss of safety functions that could lead to serious events or result in conditions that the safety systems are not able to mitigate. However, indepth studies for all non-safety-grade systems had not been performed.

To address this issue, the staff performed an indepth evaluation of the control systems that are typically used during normal plant operation. The staff has completed part of this evaluation and published the results in NUREG/CR-3958, "Effects of Control System Failures on Transients, Accidents, and Core-Melt Frequencies at a Combustion Engineering Pressurized Water Reactor" (March 31, 1986), NUREG/CR-4385, "Effects of Control System Failures on Transients, Accidents, and Core-Melt Frequencies at a Westinghouse Pressurized Water Reactor" (November 30, 1985), NUREG/CR-4386, "Effects of Control System Failures on Transients, Accidents, and Core-Melt Frequencies at a Babcock and Wilcox Pressurized Water Reactor" (December 31, 1985), and NUREG/CR-4387, "Effects of Control System Failures on Transients, Accidents, and Core-Melt Frequencies at a General Electric Pressurized Water Reactor" (December 31, 1988). In these reports, the staff identified several specific concerns with some possible solutions, including reduction of the

- potential for steam generator overfill in PWRs by including an automatic high-water-level trip for the main and emergency feedwater systems
- potential for spillover of reactor coolant into the main steamlines of BWRs by providing a safety-grade high-water-level trip of feedwater
- probability of a small-break loss-of-coolant accident (LOCA) by including automatic actuation of pilot-operated relief valve (PORV) block valves to protect against inadvertent PORV lifts
- probability of a small-break LOCA by providing increased injection capability for the safety injection system

Concerns have also been raised regarding the dc power systems, including

- the potential for a critical challenge to the standby engineered safety features (ESFs) by an instrumentation and control power supply fault
- defeat of some of the ESFs called on to mitigate an initiating event caused by the same instrumentation and control power supply fault
- the potential for the complete or partial blinding of the operators to the status of the plant by the same instrumentation and control power supply fault

These concerns have not yet been evaluated by the staff.

<u>EPRI Proposal</u>: The Evolutionary Requirements Document requires a highly reliable safety-grade control system so that non-safety control system

failures will not pose an unacceptable risk. It contains a number of specific design features to address the NRC concerns in regard to this issue, including the following:

- Adverse system interactions identified during the design or review process will be eliminated (Section 2.3.1.5, Chapter 1).
- Safety systems will be provided with independent emergency power backup (Section 2.2.8, Chapter 5).
- At least two independent connections will be made to offsite power (Section 2.2.9, Chapter 5).
- Specified safety system functions will be ensured by redundant divisions (Section 2.3.1, Chapter 5).
- Each division will be independent and separated (Section 2.3.2, Chapter 5).
- Depressurization and cooldown will be possible using safety-grade equipment, assuming an initiating event and the most limiting single failure (Section 2.3.4, Chapter 5).
- Active containment systems will be safety grade and single-failure proof (Section 2.4.1.2, Chapter 5).
- Redundant safety-grade vessel water injection will be available (Section 3.2.1, Chapter 5).
- The decay heat removal system will be redundant and safety grade (Section 3.3.1, Chapter 5).
- The diverse reactivity controls will meet the applicable requirements of GDC 26 of Appendix A to 10 CFR Part 50 (Section 3.5, Chapter 5).
- The design will include very high reliability and low failure propagation characteristics (Section 2.1.4, Chapter 10).
- The design will possess sufficient segmentation and independence characteristics (Section 2.2.3, Chapter 10).
- Automatic control system failures will be addressed (Section 2.2.9, Chapter 10).
- Component failures will be evaluated (Section 3.1.3.4, Chapter 10).
- Appropriate failure modes will be considered in the design (Section 3.5.1.1, Chapter 10).
- Common sensing instrumentation will be minimized (Section 3.5.3.1.1, Chapter 10).
- Each function will use different processors (Section 3.5.3.1.2, Chapter 10).

- Segmented equipment will be physically separated (Section 3.5.3.1.3, Chapter 10).

- Multiplexers will use separate power supplies and be housed in separate enclosures (Section 3.5.3.1.4, Chapter 10).

- The design of the isolation devices will maintain Class 1E integrity in accordance with RG 1.75, "Physical Independence of Electric Systems" (Section 6.2.6.1, Chapter 10).

- Software interfaces between safety-related and non-safety-related systems will be discouraged (Section 6.3.2.2, Chapter 10).

In addition, EPRI specifies requirements for BWR designs that the design include three independent divisions of core coolant inventory control (CCIC) and decay heat removal (DHR) systems, the CCIC and DHR divisions be separated, and the CCIC and DHR divisions be independently powered.

EPRI also specifies additional requirements for PWR designs that the design not include provisions for an automatic PORV, leak detection for each pressurizer safety valve be provided, the design include safety-grade actuation for the emergency feedwater system, the design have two independent divisions for CCIC and DHR functions, the design include two independent residual heat removal divisions, the design provide small-break LOCA coverage with the safety injection system, and the safety depressurization and vent system have two valves in series and leak detection capability.

Staff Evaluation: In its DSER for Chapter 10, the staff stated that EPRI's commitment (letter to the NRC dated December 21, 1990) to comply with GL 89-19 was acceptable and that it would confirm that this commitment was acceptably incorporated into the Evolutionary Requirements Document. Revision 2 of the Evolutionary Requirements Document indicates a commitment to comply with GL 89-19 in Table B.1-2 in Appendix B to Chapter 1, which resolves the DSER confirmatory issue. The staff also stated that the Evolutionary Requirements Document did not provide the necessary design details to be able to determine if the plant-specific design and arrangement will be adequate to resolve this issue. In addition, Issue 76 has not yet been prioritized by the staff and final resolution of this issue may dictate actions for ALWR evolutionary plant designs. Therefore, the staff will evaluate the resolution of this issue and the design details, if necessary, during its review of individual applications for FDA/DC.

3.2.29 79, Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown

Issue: The upper head region of a PWR reactor vessel (RV) is separated from the lower portion by the upper internals support plate. During power operation, "forced" reactor coolant flow is provided by the reactor coolant pumps (RCPs). For normal reactor shutdowns, RCPs are shut down sequentially as power is reduced, allowing the entire RV to cool down at a controlled rate without significant thermal gradients between the upper and lower portions of the RV. However, sudden loss of forced reactor coolant flow, such as that which occurs during a loss of offsite power, results in a natural convection cooldown (NCC) condition, with little or no cooling flow being supplied to the

upper head. Additionally, if the NCC proceeds at an unacceptably high cooling rate, substantial thermally induced stresses may occur in the RV closure flanges and studs, possibly exceeding ASME Code allowable stresses.

EPRI Proposal: EPRI recognizes the need to evaluate NCC events as part of ASME Code RV evaluations and states in Section 3.4.2.4 of Appendix B to Chapter 1 of the Evolutionary Requirements Document that an NCC event should be considered to occur at least 30 times over the 60-year design life of the RV.

Staff Evaluation: In the DSER for Chapter 4, Issue 79 was left open pending generic agency resolution. Issue 79 has since been resolved. The staff's efforts to resolve it have been based on a review of a Babcock and Wilcox (B&W) NCC analysis and the results of an NCC analysis by an NRC contractor, both of which were performed for the B&W 177 fuel assembly reactor vessel (B&W 177).

The staff noted limitations associated with the complicated analysis techniques and some calculated stresses as high as 98 percent of that allowed by the ASME Code. Therefore, no definitive conclusion could be made regarding compliance with applicable regulatory criteria for B&W 177s that might experience an NCC that was outside the bounds of the analysis assumptions or for vessels other than B&W 177s and other PWR vessels that might experience a significant NCC event in the future. However, on the basis of the analysis results and qualitative extrapolation of these results, the staff concluded the following:

- The B&W 177 is considered analyzed for NCC events that are bounded by the NCC transient profile shown in Figure 3 of NUREG-1374, "Technical Findings Related to Generic Issue 79."

- It is extremely unlikely that a single NCC event will cause the failure of any existing U.S. PWR reactor vessel, even if a cooldown rate of 100 °F/hour is exceeded.

- An NCC event that did not exceed a total cooldown of 100 °F, independent of rate, would not be expected to compromise the safety of any existing U.S. PWR reactor vessel; however, it might result in the reactor vessel being outside its documented design basis.

For the above reasons and since the Evolutionary Requirements Document does not provide sufficient commitments regarding the cooldown rate to be used in ALWR NCC analyses, the staff will evaluate this matter during its review of individual applications for FDA/DC. An applicant's proposed resolution of this issue will be acceptable if it includes a requirement to analyze an NCC event from 100-percent reactor power to cold shutdown as part of the reactor vessel ASME Code design requirements and the analysis (1) assumes 30 NCC events for the 60-year design life of the reactor vessel and (2) uses the maximum allowable cooldown rate specified by ALWR technical specifications.

In the DSER, the staff concluded that EPRI needed to recognize that the reactor pressure vessel should be capable of withstanding multiple natural circulation cooldowns. Section 3.4.2.4 of Appendix B to Chapter 1 of the Evolutionary Requirements Document adequately addresses this concern by

stating that NCC events should be considered to occur at least 30 times over the 60-year design life of the reactor vessel. Therefore, this open issue is resolved.

Issue 79 has not been generically resolved by the staff. The final agency resolution of this issue is included in Generic Letter 92-02, "Resolution of Generic Issue 79, 'Unanalyzed Reactor Vessel (PWR) Thermal Stress During Natural Convection Cooldown'." The GL did not impose any new requirements on licensees or applicants. Rather, it advised recipients (PWR licensees and applicants) of the reporting requirements delineated in 10 CFR 50.73(a)(2)(ii)(B). These reporting requirements are also applicable to ALWRs.

Specifically, for any PWR (including any PWR-type ALWR) that experiences a reportable NCC which may place the RV outside of its documented design basis, the affected licensee may be required to provide confirmation that no applicable regulatory design or fracture toughness criteria have been exceeded for the RV.

3.2.30 82, Beyond-Design-Basis Accidents in Spent Fuel Pools

Issue: A typical spent fuel storage pool with high-density storage racks can hold about five times the fuel in the core. If the pool were to be drained of water, the discharged fuel from the last two refuelings might still be "fresh" enough to melt under decay heat. Additionally, the zircaloy cladding of this fuel could be ignited during the heatup. The resulting fire, in a pool equipped with high-density storage racks, could spread to most or all of the fuel in the pool. This could cause a release of fission products from the fuel matrix.

EPRI Proposal: The Evolutionary Requirements Document contains the following requirements to decrease the likelihood of a loss of fuel pool cooling in the ALWR spent fuel pool:

- Section 4.3.2.1 of Chapter 1 requires that the spent fuel pool be a seismic Category I structure.
- Section 2.3.1.1.4.1 of Chapter 7 requires the spent fuel pool to be designed to withstand heavy load drops without pool leakage that would uncover the top of the fuel.
- Section 2.3.2.5.1 of Chapter 7 requires that the spent fuel pool be arranged to prevent cask movement over the pool.
- Section 9.3.1.2 of Chapter 8 does not permit any connections to the pool that could allow the pool to be drained below the minimum level over the spent fuel.

Staff Evaluation: In the DSER for Chapter 7, the staff concluded that the design requirements were consistent with the guidance in SRP Section 9.1.2, "Spent Fuel Storage," and were, therefore, acceptable. The staff stated, however, that although the likelihood of the complete draining of the spent fuel pool was low, the use of high-density storage racks increased the probability of a zircaloy-cladding fire as compared with the use of low-density or open-frame racks. The staff concluded that the use of low-density

storage racks was justified by a favorable value/impact ratio for new designs and recommended that EPRI make a commitment to use low-density storage racks, at least for the most recently discharged fuel. Therefore, the staff will expect the FDA/DC applicant to submit a design that uses low-density storage racks in the spent fuel pool for, as a minimum, the most recently discharged fuel.

3.2.31 83, Control Room Habitability

Issue: This issue addresses the significant discrepancies found by the staff during a survey of existing plant control rooms. These discrepancies included the inconsistency of the design, construction, and operation of the control room habitability systems with the descriptions in the licensing-basis documentation. In addition, total system testing was inadequate and some testing of the control systems was not in accordance with the technical specifications.

EPRI Proposal: Proposed elements for resolving this issued are included in Section 3.5.1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document.

Staff Evaluation: The staff has determined that the design requirements in Section 8.2.2.1 of Chapter 9 of the Evolutionary Requirements Document for the control room envelope heating, ventilating, and air conditioning (HVAC) system are consistent with the guidance in SRP Sections 6.4, "Control Room Habitability Systems," and 9.4.1, "Control Room Area Ventilation System," and are, therefore, acceptable. See Section 8.2.2 of Chapter 9 of this report. However, the requirements by themselves do not provide sufficient information to be able to determine if the plant-specific design and arrangement will be adequate. Therefore, FDA/DC applicants 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.

The open issue in the DSER for Chapter 9 concerning the design of air filtration systems is closed on the basis of the clarification in Section 8.2.1.1.8 of Chapter 9 of the Evolutionary Requirements Document that the intent was to require application of the regulatory guides and reference standards to all nuclear air treatment filtration systems, not just the filters.

The open issue in the DSER concerning charcoal filters in air filtration systems is closed on the basis of EPRI's position that the use of charcoal filters in the filtration systems will be determined by analyses and evaluations on a plant-specific basis. The staff finds that Section 8.2.1.1.22 of Chapter 9 addresses the provisions for the use of charcoal filters in HVAC systems and provides design features to process activity from normal and off-normal operation. Section 8.2.1.1.8 of Chapter 9 requires that all safety-related nuclear air treatment filtration systems be designed, fabricated, installed, and tested in accordance with RG 1.52, "Design, Testing, and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," and ANSI/ASME N509, "Nuclear Power Plant Air Cleaning Units and Components," and N510, "Testing of Nuclear Air Cleaning Systems." Also, Section 8.2.2.1.8 requires that emergency filter units (EFUs), at a minimum, consist of a prefilter, high-efficiency particulate air (HEPA)

filter, and a supply fan in accordance with ANSI/ASME N509 and RG 1.52. However, RG 1.52 needs to be met by providing each EFU with the following sequential components: (1) demister, (2) prefilter, (3) HEPA filter, (4) iodine adsorber (impregnated activated carbon), (5) ducts and valves, (6) fan, and (7) related instrumentation. Heaters or cooling coils used in conjunction with heaters should be used when the humidity is to be controlled before filtration. The downstream HEPA filter is needed to collect carbon fines from the upstream charcoal filter. The staff considers that it is incorrect to assume the regulatory guide efficiencies for the removal of elemental and organic iodine from the emergency outside air supply if there is no HEPA filter downstream of the charcoal adsorber for collecting carbon fines. The staff believes that an EFU, including a charcoal filter, should be safety related and the post-filter following a charcoal filter should be a HEPA filter to collect carbon fines. The staff will review individual applications for FDA/DC to the criteria in SRP Sections 6.4 and 9.4.1 to meet the requirements of GDC 19 of Appendix A to 10 CFR Part 50.

The open issue in the DSER concerning the control room capacity following a design-basis accident is closed because (1) Section 8.2.2.1.1 of Chapter 9 of the Evolutionary Requirements Document has been revised to state that the main control room HVAC system will be designed as a Safety Class 3, seismic Category I system; (2) the same section has been revised to change the occupancy requirement from 5 days to at least 6 days; and (3) Section 8.2.2 of Chapter 9 has been revised to include the requirement of 10 CFR 73.55(c)(6) that the control room be bullet resistant. The staff will review individual applications for FDA/DC to the criteria in SRP Sections 6.4 and 9.4.1 to meet the requirements of GDC 19 during the entire duration of postulated accidents.

Section 8.2.2 of Chapter 9 states that the control complex will include the main control room envelope, computer room, Class 1E switchgear room, Class 1E battery rooms, and HVAC equipment room. The HVAC system for the main control room will be seismic Category I, Safety Class 3, and will conform to the requirements of GDC 19 with regard to the radiation exposure guidelines and guidance of SRP Sections 6.4 and 9.5.1; RGs 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," and 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release"; and ANSI/ANS-59.2, "Safety Criteria for Nuclear Power Plant HVAC Systems Located Outside Primary Containment," for providing a safe working environment for control room operating personnel. It will provide protection against (1) any airborne hazardous chemicals above permissible concentrations, smoke in the control room, or accidental exposure of control room personnel to a high level of airborne radioactivity during emergency or isolation modes and (2) dust, smoke, and airborne radioactivity originating outside the pressure boundary during normal operation. The control room envelope will be slightly pressurized during normal operation and at least 1/8-inch water gauge will be maintained with respect to the surrounding areas during radiological conditions to prevent the entry of dust, smoke, and airborne radioactivity.

EPRI states that the control room HVAC system and other specific areas will be designed in accordance with National Fire Protection Association (NFPA) 90A, "Installation of Air Conditioning and Ventilating Systems." As stated in Section 3.3.1 of Chapter 9 of this report, by committing to NFPA 90A, EPRI has committed to use the HVAC systems for removing smoke from specific areas as a means of satisfying the smoke control provisions of NRC fire protection

guidelines. The staff will evaluate the design, installation, and operation of the HVAC systems functioning in the smoke removal mode during its review of individual applications for FDA/DC.

In Section 3.5.1.4 of Appendix B to Chapter 1, EPRI states that the resolution of Issue 83, in general, is not design related. EPRI considers the concerns of this issue to be concerns about operating plant activities in the areas of maintenance, testing, installation, and training and, therefore, beyond the scope of the Evolutionary Requirements Document. However, Issue 83 addresses significant discrepancies found during a survey of existing plant control rooms. These discrepancies included inconsistencies in the design, construction, and operation of the control room habitability systems (compared with the descriptions in the licensing-basis documentation). In addition, the staff determined that total system testing was inadequate and that some testing of the control systems was not in accordance with the technical specifications.

The staff considers EPRI's proposed resolution of Issue 83 to be complete, subject to the staff's plant-specific reviews, as applicable, of individual applications for FDA/DC. During its reviews, the staff will ensure that the existing minimum requirements, including those mentioned above, have been met, by conforming to the guidance of SRP Sections 6.5 and 9.4.1 (to meet GDC 19). Appropriate requirements are identified in the Evolutionary Requirements Document that include conformance with the applicable regulatory guides (RGs 1.52, 1.78, and 1.95) and industry standards (ANSI/ANS-59.2, ANSI/ASME N509, and ANSI/ASME N510). However, during its plant-specific reviews, the staff will also verify conformance with industry standards (ASME Code AG-1, "Code on Nuclear Air and Gas Treatment," and American Society for Testing and Materials (ASTM) D3803, "Standard Test Methods for Radiological Testing of Nuclear-Grade Gas-Phase Absorbents") and evaluation methodology and verify that technical specifications and surveillance requirements for the control room habitability system as a whole to meet GDC 4, 5, and 19 have been proposed. This will demonstrate that the control room will adequately protect the control room operators and will remain habitable in accordance with TMI Task Action Plan Item III.D.3.4 (NUREG-0737, "Clarification of TMI Action Plan Requirements"). Additionally, the FDA/DC applicant will have to ensure that the control room habitability design meets GDC 4, 5, and 19 and that operators are protected in accordance with TMI Action Plan Item III.D.3.4. The combined license applicant will have to verify (1) that the as-built design and the operating, maintenance, and emergency procedures and training and the performance characteristics of the control room habitability system are consistent with the licensing-basis documentation and (2) that the technical specifications and surveillance procedures are consistent with the licensing-basis documentation. The combined license applicant will also have to provide adequate verification of system performance and integrity.

A possible resolution of Issue 83 has been identified. This resolution is documented in a staff report (H. Denton to W. Dircks, Executive Director for Operations) dated June 19, 1984, and a contractor report (T. Powers, Pacific Northwest Laboratory, to W. Milstead, NRC) dated December 3, 1984, both on control room habitability. Efforts included a survey of 11 plants and the issuance of several other contractor reports. Current efforts include preparation of a generic letter and SRP revisions. Should final agency

resolution of this issue dictate additional actions for ALWR evolutionary plant designs, the staff will expect the FDA/DC applicant to address the resolution of Issue 83 for staff evaluation.

3.2.32 84, Combustion Engineering Power-Operated Relief Valves

Issue: Following the TMI-2 accident, the purpose and use of power-operated relief valves (PORVs) was the subject of considerable analyses and discussion. Although PORVs were originally installed to prevent challenges to the spring-operated safety valves, plants designed by Westinghouse and Babcock and Wilcox (B&W) sometimes rely on PORVs for depressurization during certain design-basis events (e.g., steam generator tube rupture). Another use of PORVs at some plants is to provide low-temperature overpressure protection.

When this issue was being evaluated, all Westinghouse and B&W and older Combustion Engineering (CE) PWRs had PORVs. However, the newer CE designs did not include PORVs. The staff's review of this issue in 1984 showed that existing CE plants without PORVs met regulatory requirements, but that other considerations, primarily accident management for beyond-design-basis events and potential core-melt risk reduction, required further study. However, the staff considered events for which PORVs could prove to be of benefit to have a low probability of occurrence and was unaware of any immediate safety concerns associated with the absence of PORVs in CE-designed plants.

EPRI Proposal: Proposed elements for resolving this issue are included in Section 3.5.4 of Appendix B to Chapter 1 of the Evolutionary Requirements Document. PWR ALWRs will not have PORVs. Instead, the Evolutionary Requirements Document calls for a single-failure-proof safety depressurization and vent system (SDVS). The SDVS will be a dedicated safety-grade system that will be capable of performing those safety-related functions, with the exception of the low-temperature overpressure protection function, that can be performed by PORVs in the current generation of PWRs.

Staff Evaluation: In its DSER for Chapter 5, the staff concluded that the Evolutionary Requirements Document needed to specify a criterion for the depressurization rate of the SDVS. In Section 6.6.4.3 of Chapter 5, EPRI specifies that the SDVS will be capable of depressurizing the reactor coolant system below 250 psig before reactor vessel melt-through as a means of precluding containment challenges through direct containment heating. In Section 5.4 of Chapter 5 of this report, the staff concludes that this is an acceptable design objective. Therefore, this open issue is closed.

The staff has resolved Issue 84 and has established no new requirements. As discussed in Section 3.1 of this appendix, Issue 84 is not included in Appendix B to NUREG-0933.

3.2.33 87, Failure of High-Pressure Coolant Injection Steamline Without Isolation

Issue: This issue addresses a postulated break in the high-pressure coolant injection (HPCI) steam supply line and the uncertainty regarding the operability of the HPCI steam supply line isolation valves under those conditions. A similar situation can occur in the reactor water cleanup (RWCU) system.

The HPCI steam supply line has two containment isolation valves in series (one inside the containment and one outside), both of which are normally open in most plants (two plants do operate with the outboard isolation valve normally closed). An HPCI supply valve located adjacent to the turbine and the turbine stop valve are normally closed. The RWCU system also has two normally open containment isolation valves that must remain open if the system is to function.

At the valve manufacturers' facilities, only the opening characteristics are tested under operating conditions (because of flow limitations). Although the operation of the valves is tested periodically without steam, the capability of the valves to close against the forces created by the steam flow resulting from a downstream line break has not been demonstrated.

The valve type is not under General Electric's (BWR vendor) scope of control, but is selected by the plant architect-engineer. This results in a diversity of valves and valve types (Y-type globe valves and gate valves) and increases the difficultly of demonstrating valve operating capability.

EPRI Proposal: In Section 3.5.5.5 of Appendix B to Chapter 1 of the Evolutionary Requirements Document, EPRI asserts that the motor-operated valve (MOV) testing and surveillance issue raised by Issue 87 should be considered resolved for the ALWR evolutionary plant designs. EPRI gives two arguments as a basis for this statement.

First, EPRI asserts that the NRC concerns about the performance of safety-related MOVs under design-basis conditions are addressed by the specific requirements sections of the Evolutionary Requirements Document. Second, EPRI states that it will take full advantage of its program to develop an MOV performance prediction methodology.

Staff Evaluation: The staff has resolved Issue 87; it issued a closeout memorandum (E. Beckjord to J. Taylor) on December 9, 1991.

The staff does not agree with EPRI's position that the MOV issue should be considered resolved for the ALWR evolutionary plant. For example, as discussed in Section 12.2.2 of Chapter 1 of this report, the specific requirements sections of the Evolutionary Requirements Document have not fully resolved NRC staff concerns regarding the performance of MOVs in the ALWR evolutionary plant. Also, EPRI's MOV performance prediction program is in its initial stages and cannot be relied on until successful results begin to be obtained. Finally, the ALWR evolutionary plant designs contain a few highly important MOVs (such as the RWCU system isolation MOVs) that would need to function properly in the event of an abnormal plant condition.

The staff will evaluate this matter during its review of individual applications for FDA/DC in accordance with the above position and its positions on MOVs stated in Section 12.2.2 of Chapter 1 of this report.

3.2.34 94, Additional Low-Temperature Overpressure Protection for Light-Water Reactors

Issue: Low-temperature overpressurization was originally identified as a safety issue in November 1978. The resolution of this safety issue was given in a revision to SRP Section 5.2, "Overpressure Protection." However, from

1979 to July 1983, 12 pressure transients were reported to the NRC. Of these 12 pressure transients, 2 overpressurization transients suggested potential weaknesses in the existing overpressure protection criterion or in its implementation. Specifically, these two overpressurization transients suggested weaknesses relating to the operability and reliability of both trains of low-temperature overpressure protection when the plant was in a solid (water solid) condition.

Major overpressurization of the reactor coolant system, if combined with a critical-size crack, could result in a brittle failure of the reactor vessel. Failure of the reactor vessel could make it impossible to provide adequate coolant to the reactor and result in major core damage or a core-melt accident. This issue mainly affects the design and operation of PWRs.

EPRI Proposal: In Section 3.3.3 of Appendix B to Chapter 1 of the Evolutionary Requirements Document, EPRI discusses low-temperature overpressure protection (LTOP) for ALWRs. EPRI's design objective is to minimize the need for any special LTOP, while providing the capability for reliable LTOP should it be required. To accomplish this objective, EPRI states that primary system components are to be made of materials with a low nil ductility transition temperature. In addition, adequate relief capacity is to be provided for LTOP.

Section 2.2.2 of Chapter 4 of the Evolutionary Requirements Document specifies that materials, design, and fabrication methods for the reactor pressure vessel are to selected so that the materials will not behave in a brittle manner, be subject to rapidly propagating failure, or otherwise fail under normal operations and specified plant transients or events. Section 2.3.1.2 of Chapter 4 requires that the reactor pressure vessel base material and welding material chemistry be controlled so that material susceptibility to neutron embrittlement is as low as practicable. Section 3.3.2 of Chapter 3 provides requirements for LTOP in PWRs. Section 5.2.3.3.1 of Chapter 5 requires that relief valves be provided to ensure that the residual heat removal design pressure will not be exceeded as a result of the most limiting event. In addition, guidelines for relief capacity for LTOP are specified in Section 5.2.3.3 of Chapter 5.

Staff Evaluation: In the DSERs for Chapters 3 and 4, the staff noted that EPRI's additional requirements for LTOP for ALWRs appeared to be adequate and reasonable. However, the staff was unable to make a final judgment on the adequacy of EPRI's LTOP provisions because it was considering new requirements as part of the resolution of Issue 94. Therefore, this was identified as an open issue in both DSERs.

The staff has established guidance for resolving the LTOP concerns for operating reactors in NUREG-1326, "Regulatory Analysis for the Resolution of Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,'" but has not yet finalized its position for ALWRs. Therefore, it will evaluate the adequacy of the specific design for LTOP during its review of individual applications for FDA/DC.

3.2.35 96, Residual Heat Removal Suction Valve Testing (integrated into Issue 105)

Issue 96 has been subsumed by Issue 105. See Section 3.2.39 for the staff's evaluation of Issue 105.

3.2.36 99, Reactor Coolant System/Residual Heat Removal Suction Line Valve Interlock in PWRs

According to Appendix B to NUREG-0933, Issue 99 is not applicable to future plants. See the staff's evaluation of Issue 105 in Section 3.2.39.

3.2.37 101, BWR Water Level Redundancy

Issue: This issue concerned the potential for a postulated break in an instrument line of a BWR in conjunction with the worst-case single failure. Should one of two reference columns break, a single failure associated with the other reference column could completely defeat mitigation systems for the resulting transients. BWRs typically have two reference columns in the water level instrumentation. Water level is measured by means of differential pressure sensors connected between the reactor vessel and the reference columns, which are full of water. Should a reference column break, the water in it would flash to steam and the water level indication in all channels connected to the broken column would give a false "high" reading. Should one of the reference columns break and should a worst-case single failure cause the other column to indicate high water level, the false signals could result in a low-water-level situation that could prevent mitigation systems, such as the emergency core cooling system, from initiating. In some plants, this specific scenario could lead to core uncovery. However, because there are several designs among the operating plants, there is not a single, generic solution to resolve the problem.

EPRI Proposal: Chapter 10 of the Evolutionary Requirements Document lists requirements for the design process and on reliability that should ensure that the designs complying with the Evolutionary Requirements Document will not be subjected to the postulated scenario. Section 3.1.3.1 of Chapter 10 requires that the man-machine interface system (M-MIS) designer review existing LWR plant designs to identify problems and develop their solutions. EPRI states that the requirements for availability and reliability of Section 3.5.1.2 will ensure that no single random failure of any M-MIS equipment will cause a forced outage. Sections 3.5.3 and 3.5.4.3 address multiple random failures of M-MIS equipment. In addition, Section 3.3.3.2.1 of Chapter 4 requires four divisions of wide-range level instruments with four spatially separated taps. EPRI considers that implementation of these requirements resolves this issue.

Staff Evaluation: In its DSER for Chapter 10, the staff concluded that EPRI's commitment (letter to the NRC dated December 21, 1990) to include compliance with the resolution of Issue 101 in accordance with GL 89-11, "Resolution of Generic Issue 101, 'Boiling Water Reactor Water Level Redundancy,'" in the Evolutionary Requirements Document was acceptable. The staff has confirmed that Table B.1-2 of Appendix B to Chapter 1 of the Evolutionary Requirements Document indicates a commitment to comply with GL 89-11; therefore, this confirmatory issue is resolved. However, the Evolutionary Requirements

Document does not provide sufficient information for the staff to complete its review of this issue. The staff will evaluate resolution of Issue 101 during its review of individual applications for FDA/DC.

3.2.38 103, Design for Probable Maximum Precipitation

Issue: Lack of adequate drainage at reactor sites may render safety-related equipment inoperable. New procedures for estimating the probable maximum precipitation (PMP) result in a more severe design-basis probable maximum flood than that determined using previous methodologies. In 1989, the staff revised SRP Sections 2.4.2, "Floods," and 2.4.3, "Probable Maximum Flood (PMF) on Streams and Rivers" (Revision 3) to incorporate the PMP procedures and criteria contained in the latest National Weather Service publications. In addition, on October 19, 1989, the staff issued GL 89-22, "Potential for Increased Roof Loads and Plant Area Flood Runoff Depth at Licensed Nuclear Plants Due to Recent Change in Probable Maximum Precipitation Criteria Developed by the National Weather Service," to inform the industry of these revisions.

EPRI Proposal: Table B.1-2 in the Evolutionary Requirements Document indicates that the ALWR design will meet SRP Sections 2.4.2 and 2.4.3 (Revision 3) and GL 89-22.

Staff Evaluation: Section 3.3.2 of Chapter 6 of the Evolutionary Requirements Document (Revision 0) referenced ANSI/ANS-2.8-1981 as part of the resolution of this issue at the time the DSER for Chapter 6 was issued. At that time, the staff had not reviewed this standard for consistency with regulatory requirements and staff guidance. Consequently, the staff did not complete its review of Issue 103 in the DSER and left review of this issue open.

On the basis of the commitment in the current version of the Evolutionary Requirements Document that the ALWR design will meet SRP Sections 2.4.2 and 2.4.3 (Revision 3) and GL 89-22, the staff concludes that EPRI has adequately addressed the resolution of Issue 103.

3.2.39 105, Interfacing Systems Loss-of-Coolant Accident at LWRs (subsumes Issue 96)

Issue: Either human errors, valve leakage, multiple valve failures, or a combination of these could permit overpressurization of low-pressure systems connected to high-pressure systems. Pipe or component rupture resulting from overpressurization could result in loss of decay heat removal if the rupture was outside the containment.

EPRI Proposal: Section 3.5.2 of Appendix B to Chapter 1 of the Evolutionary Requirements Document states that "the ALWR is being required to satisfy all of the current regulatory requirements which are intended to assure an acceptable level of isolation capability." This section also indicates that all ALWRs are required to satisfy the following additional requirements:

- All systems and subsystems connected to the reactor coolant system (RCS) that extend outside the primary containment boundary will be designed to the extent practicable to an ultimate rupture strength (URS) at least equal to full RCS pressure.

- For those interfacing systems or subsystems that do not meet the full reactor coolant system URS requirement, the plant designer will determine by evaluation if the degree and quality of isolation or reduced severity of the potential pressure challenges are low enough to preclude an intersystem loss-of-coolant accident (ISLOCA).

- Each of the high- to low-pressure interfaces (viz., those between the systems normally exposed to the RCS pressure and the interfacing lower pressure system) will include the following protective measures:

  - the capability for leak testing of the pressure isolation valves

  - valve position indication that is available in the main control room even when isolation valve operators are deenergized

  - 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

Additional requirements for BWRs are the following:

- An inboard testable check valve and an outboard motor-operated valve are to be used on the decay heat removal (DHR) injection lines. High-pressure piping and valves are to be used through the outboard isolation valve. Pressure interlocks are to protect low-pressure piping upstream of the outboard isolation valve.

- Check valves will be testable to verify free movement of the valve disk.

- Relief valves are to be provided in the low-pressure pump discharge piping to protect against overpressure due to backleakage from the RCS.

- The portions of the DHR system that connect directly to the RCS will be designed to a URS at least equal to full RCS pressure.

- Interlocks are to be provided as follows:

  - a two-way interlock so that it is not possible to have an open shutdown connection to the vessel in any given loop whenever the pool suction, pool discharge valve, or wetwell spray valves are open in the same loop

  - redundant interlocks to prevent the opening of the shutdown connections to and from the vessel whenever the pressure is above the shutdown range with increasing pressure causing closure of these valves

  - redundant interlocks preventing the opening or closing of the shutdown suction connections to the vessel in any given loop and discharge to radioactive waste whenever a low-reactor-water-level signal is present

Additional requirements for PWRs are the following:

- The residual heat removal (RHR) system will be designed for a pressure of 900 psig and a temperature of 400 °F and will use a minimum of Schedule 40 piping.
- Relief valves will be provided to ensure that RHR system design pressure will not be exceeded as a result of the most limiting event. The plant designer is to define the most limiting event on the basis of a review of experience with overpressure transients during RHR operation, including such events as inadvertent startup of the safety injection system and startup of reactor coolant pumps with coolant in the steam generators that is above average RCS temperature.

- Interlocks will be provided for the RHR suction isolation valves to prevent the valves from opening in the event RCS pressure exceeds RHR design pressure. An interlock that automatically closes the isolation valves on high pressure is not to be provided.

- Portions of the RHR system that connect directly to the RCS will be designed to ensure that the URS will not be exceeded at full RCS pressure.

Staff Evaluation: The staff's DSER for Chapter 5 contained an open issue relating to low-pressure systems not designed to withstand full RCS pressure. The staff recommended that evolutionary ALWRs provide

- the capability for leak testing of the pressure isolation valves

- valve position indication that is available in the control room when isolation valve operators are deenergized

- high-pressure alarms to warn control room operators when rising RCS pressure approaches the design pressure of attached low-pressure systems and both isolation valves are not closed

This recommendation is reflected by requirements in Section 2.2.14.3 of Chapter 3 of the Evolutionary Requirements Document; therefore, this open issue is resolved.

SECY-90-016 discusses the ISLOCA, focusing on a URS of systems and subsystems connected to the RCS that is at least equal to the full RCS pressure as a way of reducing the probability of an ISLOCA outside the containment. In SECY-90-016, the staff concludes that designing low-pressure systems to withstand full RCS pressure, to the extent practicable, is an acceptable means of resolving the ISLOCA issue. (Section 2.2.14.1 of Chapter 3 of the Evolutionary Requirements Document provides a requirement to this effect.) Nevertheless, the staff considers alternative design approaches, such as inclusion of pressure relief valves in low-pressure system piping, acceptable in principle, provided due consideration is given to relief capacity, effluent conservation, possible defeat of pressure isolation valve interlocks, and relief valve reliability. Also in this regard, the staff noted in SECY-90-016 that for some low-pressure systems attached to the RCS, it might not be practical or necessary to provide a higher system ultimate pressure capability for the entire low-pressure system (this is also noted in the DSER). However, uniform criteria and design stresses allowable for ISLOCA conditions in low-

pressure piping systems have yet to be developed and approved by the staff for ALWRs, although ASME has been asked to consider developing Boiler and Pressure Vessel Code requirements for ISLOCA design in future reactors. In the meantime, the staff will evaluate exceptions on a case-by-case basis during its review of individual applications for FDA/DC.

In a staff requirements memorandum (SRM) dated June 26, 1990, the Commission approved the staff's position in SECY-90-016 on ISLOCA, provided all elements of the low-pressure system are considered (e.g., instrument lines, pump seals, heat exchanger tubes, and valve bonnets).

The staff concludes that the Evolutionary Requirements Document contains requirements that are sufficient to resolve this issue. However, since the document does not specifically identify interfacing systems other than the DHR or RHR systems, the staff will evaluate other interfacing systems designs during its review of individual applications for FDA/DC. It will ensure that all elements of low-pressure systems have been considered in accordance with the SRM of June 26, 1990, particularly those to be evaluated under Section 2.2.14.2 of Chapter 3 of the Evolutionary Requirements Document.

As noted in the DSER, for additional assurance that low-pressure systems will not be exposed to RCS pressure, the staff will review inservice testing programs and technical specifications for ALWR applications to ensure that the appropriate pressure isolation valves are identified and adequately tested.

Issue 105 has not yet been generically resolved by the staff and has a high safety priority. Should the final resolution of this issue indicate additional actions for ALWR evolutionary plant designs, the staff will expect the FDA/DC applicant to address the resolution of Issue 105 for staff evaluation.

3.2.40 106, Piping and Use of Highly Combustible Gases in Vital Areas

Issue: Issue 106 addresses the risk associated with the use of hydrogen and other combustible gases, such as propane and acetylene, during normal plant operation. It does not cover the use of large quantities of liquid hydrogen at hydrogen water chemistry (HWC) installations at BWRs or liquified petroleum gases that are covered under Licensing Issue 136.

EPRI Proposal: In the note to Table B.3-1 in Appendix B to Chapter 1 of the Evolutionary Requirements Document, EPRI states that Issue 106 was determined, through the NUREG-1197 screening process, to be not technically relevant to the ALWR design.

Table B.1-2 in Appendix B to Chapter 1 of the Evolutionary Requirements Document shows an ALWR commitment to comply with SRP Section 9.5.1, Revision 3, "Fire Protection Program."

Staff Evaluation: In 1987, the staff accepted licensing topical report EPRI NP-5283-SR-A, "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations," which provides guidelines for HWC installations. The guidelines give a number of system design features and administrative controls that are in addition to, or more restrictive than, those in SRP Section 9.5.1. These include

- new relations for separation distances for gaseous and liquid hydrogen storage facilities

- use of color-coded piping and warning signs (ANSI A13.1, "Scheme for the Identification of Piping Systems," and ANSI Z35.1, "Specification for Accident Prevention Signs")

- excess flow valves, system trips, and other design features (e.g., hydrogen detectors) to help mitigate the consequences of leaks or breaks and to perform the intended design function with or without normal ventilation, and as a minimum, a system trouble alarm and/or annunciator in the main control room

- periodic retesting to verify operability and functional performance of equipment

In its letter transmitting the safety evaluation report (SER) on Report NP-5283-SR-A to EPRI, the staff recommended that the report be modified to include hydrogen systems supplying hydrogen to the volume control tank at PWRs and for cooling electric generators at PWRs and BWRs. With the exception of information dealing specifically with the HWC application (e.g., certain trips, injection points, main steamline radiation), most of the EPRI guidelines dealing with hydrogen are applicable to these other uses. Although EPRI did not modify the report as suggested by the staff, the staff used the information in this report in evaluating Issue 106 for evolutionary ALWRs.

The Evolutionary Requirements Document does not address the reasons for determining that this issue is not technically relevant and does not provide sufficient information to enable the staff to fully evaluate EPRI's proposed requirements related to resolution of this issue. Consequently, the staff will evaluate the resolution of Issue 106 during its review of individual applications for FDA/DC. This issue may be resolved with respect to the volume control tank and generator applications by an applicant's commitment to comply with

- the current SRP Section 9.5.1, Branch Technical Position (BTP) CMEB 9.5-1, Part C.5.d(5), modified as shown below

- the staff's SER and the guidelines in EPRI NP-5283-SR-A pertaining to hydrogen that do not deal specifically with the HWC application

- the following general guidance:

  - BTP CMEB 9.5-1, Part C.5.d(5), modified as follows: "Hydrogen lines in safety-related areas should follow the guidance of Regulatory Guide 1.29, 'Seismic Design Classification,' Section C.2. The lines should (1) be equipped with an excess flow valve or equivalent protection located outside the building so that in case of a line break, the hydrogen concentration in affected areas does not exceed 2 percent volume or (2) be sleeved with the outer pipe vented directly to the outside."

  - The hydrogen system piping and components should be located to reduce risk (e.g., by reducing piping length and proximity to safety-related equipment).

- Design features (e.g., removable spool pieces) and/or administrative controls should be provided to prevent inadvertent bypass of small or normally isolated hydrogen supplies, or flow-limiting devices should be used to limit hydrogen releases to a leak or break.

- Equipment and controls to mitigate the consequences of a hydrogen fire or explosion should be accessible and remain functional during an event.

- Design features and/or administrative controls should be provided to isolate the hydrogen supply if normal building ventilation is lost (e.g., as a result of building isolation caused by operator actions following a hydrogen fire or explosion in an auxiliary building).

- Backflow to a leak or break of hydrogen contained in components (e.g., generator) should be considered in evaluating the consequences of leaks or breaks, and measures should be taken to mitigate these consequences.

- Threaded joints in the hydrogen distribution lines within safety-related areas should be back welded.

- Safety-related equipment should not be located in the turbine building because of the hazards associated with hydrogen fires or explosions and large oil fires and the large uncertainties in estimating the consequences.

Issue 106 has not yet been generically resolved by the staff and has a medium safety priority. Should the final resolution of this issue indicate additional actions for ALWR evolutionary plant designs, the staff will expect the FDA/DC applicant to address the resolution of Issue 106 for staff evaluation.

3.2.41 107, Generic Implications of Main Transformer Failures

Issue: As a result of main transformer faults at the North Anna Power Station, generic concerns were raised concerning the suppression of transformer fires and the impact of these fires on plant safety systems. A transformer fire of sufficient magnitude (location dependent) has the potential for degrading plant safety equipment and safety systems. The generic concerns arising from this issue involve (1) the proper maintenance, storage, and handling of transformers to prevent transformer failure and (2) the mitigation and containment of transformer fires. The staff has determined that there are four key areas in the prevention and control of transformer fires that should be addressed. These are the deluge system, drainage system, fire barriers, and firefighting and related procedures. Other generic concerns involve the layout and segregation of the transformer bay drains, use of fire barriers, and hindrances to firefighting related to access, communications, mobility, training, and procedures.

EPRI Proposal: Chapters 1, 6, and 9 of the Evolutionary Requirements Document provide design requirements for fire protection systems and include the following specific design criteria to address the safety concerns of this issue:

- Chapter 1 includes a general requirement that the evolutionary ALWR plant design provide for an integrated approach to fire prevention and mitigation of fire damage.

- Section 2.3.3.11 of Chapter 6 requires that outdoor oil-filled transformers have oil spill confinement features or drainage away from the buildings. Such transformers are to be located at least 50 feet from the building, or if building walls are within 50 feet of the oil-filled transformers, these walls will not have openings and will have a fire-resistance rating of at least 3 hours. The outdoor oil-filled transformers are to be protected by deluge systems and drainage to accommodate the flow of oil and water as determined by the fire hazard analysis.

- Section 3.3.3.1 of Chapter 9 requires that the turbine-generator and associated areas described in NFPA 803, "Fire Protection for Light Water Nuclear Power Plants," be protected in accordance with NFPA 803.

- Section 3.3.3.3 of Chapter 9 specifies that automatic fixed water suppression over the fire area be provided for any equipment identified by the fire hazard analysis as containing a sufficient quantity of combustible material to warrant a fixed suppression system.

Staff Evaluation: In its DSER for Chapter 11, the staff stated that Section 2.3 of the DSER for Chapter 6 described an open issue regarding the location of oil-filled transformers in relation to exterior building walls. Section 2.3.3.11 of Chapter 6 of the Evolutionary Requirements Document provides requirements for controlling transformer fires that may damage the nearby buildings in an ALWR plant that the staff finds acceptable (see Section 2.3 of Chapter 6 of this report). Therefore, this open issue is resolved.

3.2.42 110, Equipment Protective Devices on Engineered Safety Features

Issue: Essential equipment has experienced a large number of failures or has been incapacitated as a result of either the failure or the intentional bypass of protective devices intended to trip active engineered safety features (ESFs) for indications of equipment faults. The affected systems exist throughout the plant and include the plant control system, the plant protection system, and the ESFs. The staff is concerned that the reliability estimates for essential equipment may not properly account for failure of the protective devices. Because of the loss of redundant devices through failures of circuits intended to be independent, there is an increased probability of common-mode failure of redundant vital services. This issue needs to be studied further to determine if failure rate estimates for essential equipment have increased and if essential equipment could be made significantly more reliable by improving the reliability of protective devices.

EPRI Proposal: The Evolutionary Requirements Document includes a number of specific design requirements to address the concerns of this issue. Chapter 5 requires that

- plant systems embody sufficient robustness of design to tolerate a conservative number of spurious or inadvertent engineered safety system actuations without the need for followup tests or inspections to verify systems' integrity or operability (Section 2.2.7)

- each division of engineered safety systems be totally independent electrically unless it is otherwise physically impracticable or less safe (Section 2.3.2)

Chapter 10 requires that:

- a robust system design, including segmentation of major functions, separation of redundant equipment within a segment, and fault-tolerant equipment, be developed to achieve high reliability and protection against the propagation of faults (Section 2.2.3)

- the man-machine interface system (M-MIS) equipment be designed and configured to readily support inservice testing by providing built-in test features, incorporating good human factors principles, and avoiding the use of undesirable features such as test jumpers or lifting of leads (Section 2.2.4)

- the M-MIS design process explicitly consider the potential for and the consequences of failures of plant and M-MIS components (Section 3.1.3.4)

- the capability for continuous on-line self-testing of hardware integrity be provided for as much of the M-MIS as is practicable (Section 3.6.1)

- on detection of a failure in the M-MIS, a system be designed so that it can be placed in such configuration that an additional single failure will not prevent system-level protection or safety action (Section 3.6.4).

- the mean time to detect and repair failures down to the lowest replaceable module, averaged across all types of M-MIS equipment for the entire design life, be less than 4 hours and the maximum time for any module be less than 8 hours (Section 3.7.4)

Staff Evaluation: In its DSER for Chapter 10, the staff agreed that the implementation of the design criteria in the Evolutionary Requirements Document would resolve this issue, with the exception of the staff's concern about lack of specific guidelines for exceptions to the independence criteria for electrical systems. Since it has not prioritized Issue 110, the staff will defer further evaluation of the open issue in the DSER to its review of individual applications for FDA/DC.

3.2.43 113, Dynamic Qualification Testing of Large-Bore Hydraulic Snubbers

Issue: Large-bore hydraulic snubbers (LBHSs) are active mechanical devices used to restrain safety-related piping and equipment during seismic or other dynamic events (e.g., high-energy line break), yet they also allow sufficient piping and component flexibility to accommodate system expansion and contraction due to thermal transients, such as normal plant heatups and cooldowns. Dynamic qualification testing and periodic functional testing are important to verify that the LBHSs are properly designed and maintained for the life of the plant. The ALWR plant design is expected to be fully optimized with respect to the use of LBHSs; that is, it is expected that as few as possible will be used and that there will be no redundancy.

EPRI Proposal: In Section 3.3.4.4.2 of Appendix B to Chapter 1 of the Evolutionary Requirements Document, EPRI recognizes that the ALWR will not make as extensive use of LBHSs as do some current plants. Furthermore, this section states: "Section 4.2.3.5 of Chapter 6 of the ALWR Utility Requirements Document requires that test requirements for hydraulic snubbers shall be established by the plant designer for their qualification for use in the ALWR." (In the elements of resolution, EPRI only commits generally to develop dynamic tests for LBHSs.)

Staff Evaluation: In its DSER for Chapter 3, the staff concluded that EPRI's commitment (in its topic paper dated July 9, 1987) that dynamic test requirements for the LBHSs will be consistent with the staff's observations on dynamic qualification testing listed in the paper was adequate to consider Issue 113 resolved for the evolutionary plants. However, this specific commitment is not contained in the current version of the Evolutionary Requirements Document.

The staff's preliminary recommendations on the resolution of this issue at this time include numerous recommendations for improving dynamic qualification and testing in order to improve the reliability of LBHSs in both existing and future plants. Additionally, an adjunct issue regarding snubber single failures is also being evaluated for application to ALWRs where the use of snubbers is expected to be reduced as much as possible.

For the above reasons and since the Evolutionary Requirements Document does not provide sufficient detailed commitments regarding this issue, if snubbers (including LBHSs) are used in ALWR evolutionary plant designs, the staff will evaluate the environmental (including dynamic) qualification and inservice inspection and testing requirements for conformance to NRC requirements that are current at the time of the application during its review of individual applications for FDA/DC.

Issue 113 has not yet been generically resolved by the staff and has a high safety priority. Should the final agency resolution of this issue dictate additional actions for ALWR evolutionary plant designs, the staff will expect the FDA/DC applicant to address the resolution of Issue 113 for staff evaluation.

3.2.44 118, Tendon Anchorage Failure

Issue: An inspection of a PWR prestressed concrete containment structure showed that three lower vertical tendon anchor heads were broken. The failures appeared to be caused by stress corrosion cracking. Quantities of water ranging from a few ounces to about 1.5 gallons were found in grease caps.

EPRI Proposal: Chapter 6 of the Evolutionary Requirements Document states that a freestanding steel containment for PWRs (Section 4.3.4.1) or a reinforced-concrete containment for BWRs (Section 4.3.3.2) will be used. Since a prestressed concrete containment is not specified, no specific requirements for tendon anchorage are provided.

Staff Evaluation: Since the Evolutionary Requirements Document indicates that prestressed concrete containments will not be used for evolutionary plant designs, this issue is not applicable. However, Issue 118 has not yet been

prioritized by the staff, and should prestressed concrete containments be used in any evolutionary plant designs, the staff will evaluate resolution of this issue, as applicable, during its review of individual applications for FDA/DC.

3.2.45 120, On-Line Testability of Protection Systems

Issue: During its 1985 review of several plant technical specifications, the staff discovered that the design of protection systems of some plants did not provide as complete a degree of on-line (at-power) surveillance testing capability as other plants undergoing staff evaluation at that time. This raised questions about the on-line testability of protection systems and the possibility that some nuclear power plants might not provide complete testing capability. Issue 120 was established to examine these questions. Protection systems consist of the reactor protection system (RPS) and the engineered safety features actuation system (ESFAS). The main concern of Issue 120, however, is the on-line testability of the actuation subgroup (slave) relays in the ESFAS.

The requirements for at-power testability of components are included in GDC 21 of Appendix A to 10 CFR Part 50. RGs 1.22, "Periodic Testing of Protection System Actuation Functions," and 1.118, "Periodic Testing of Electric Power and Protection Systems," and IEEE 338-1977, "Criteria for the Periodic Testing of Nuclear Power Generating Station Safety Systems," provide supplementary guidance. This guidance is intended to ensure that protection (including logic, actuation devices, and associated actuated equipment) will be designed to permit testing while a plant is operating at power without adversely affecting the plant's operation.

EPRI Proposal: Appendix B to Chapter 1 of the Evolutionary Requirements Document does not address Issue 120. In a letter to the NRC dated November 15, 1991, EPRI stated that Issue 120 was prioritized by the staff after the January 1, 1990, cutoff date for consideration in Appendix B to Chapter 1. EPRI further stated that Section 3.6 of Chapter 10 provides extensive requirements that appear to adequately address the technical concerns of this issue.

Chapter 10 of the Evolutionary Requirements Document addresses the testability requirements (Section 3.6) and the reactor protection and safety systems man-machine interface system (M-MIS) requirements (Section 8) as follows:

- Section 3.6.1 requires that the capability for continuous on-line self-testing of hardware integrity be provided for as much of the M-MIS as is practical. This testing should not affect the system functionality and should be performed on the module, as opposed to on-the-system basis. These tests may include, but are not limited to, RAM (random access memory) and ROM (read-only memory) failure checks, arithmetic processing unit failure checks, data-link buffer checks, and CPU (central processing unit) reset of watch-dog timers.

- Section 3.6.2 requires that the capability for periodic functional testing of the systems be provided. This periodic testing should be manually initiated, but automatically performed once initiated, and should meet the requirements of RGs 1.22 and 1.118 and IEEE 338. Automatic initiation of periodic testing may be provided where the testing does not degrade the system functionality.

- Section 3.6.7 requires that built-in, automated test features be provided for periodic functional testing, as necessary, to eliminate physical reconfiguration of systems (e.g., adding jumpers, lifting leads, swapping cables) to perform the required tests. However, initiation of periodic, automated functional tests should meet the requirements in Section 3.6.2.

- Section 3.6.8 requires the safety-related systems (e.g., reactor protection and safety systems) to have automatic test features that are sufficient to meet the technical specification requirements for periodic surveillance of the system's functionality as defined by RGs 1.22 and 1.118 and IEEE 338.

- Section 3.6.9 requires that test features of the M-MIS be designed so that, to the degree practical, the tests can be performed with the plant at power without causing spurious actuation of reactor trip devices or safety system components. Where testing at power would upset plant operation or damage equipment, provisions should be made to test the equipment with the plant operating at reduced power or in a shutdown condition. All tests required to be performed to keep the plant at power or increase power should be capable of being performed without shutting down or reducing power.

- Section 8.2.3.4 requires that the M-MIS for the protection and safety systems normally provide for testing to be initiated at the direction of the operator, but the testing itself be largely automatic. For portions of the M-MIS of these systems that can be tested without activation of system components, the incorporation of automatic testing should be evaluated by the M-MIS designer. Where automatic testing is impractical, the M-MIS designer should provide operator aids or other features to minimize the potential for operator errors. In no case should the testing impair the capability to carry out the protection or safety function. Where practical, the system should automatically realign itself when its action is called for while it is being tested.

- Section 8.3.2.3 requires that the RPS provide for automatic self-testing of as much of the system as is practical; that is, the RPS should require initiation of testing by the operators only where automatic testing is not practical.

Staff Evaluation: The positions in Chapter 10 of the Evolutionary Requirements Document regarding the testability of protection systems appear, in general, to be consistent with the staff's interim position on the resolution of Issue 120. The staff, however, has a few concerns regarding the EPRI proposals.

In several places in Chapter 10 of the Evolutionary Requirements Document, EPRI uses the wording "as much as practical." Since "as much as practical" is subject to wide interpretation, more precise wording should be chosen to avoid confusion and provide for extensive on-line testability.

On the basis of its review of Chapter 10 of the Evolutionary Requirements Document, the staff is under the impression that there is a high likelihood that the ALWR designer may choose to design the reactor protection and safety systems instrumentation and control circuits to contain sensors, all digital logic circuitry, and the combination of microprocessors and final actuation

contacts that take the place of subgroup (slave) relays (the final actuation contacts must change state in order to actuate the final actuated equipment). Should this be the case, the concern of Issue 120 regarding the on-line testability of the slave relays for the current plants is changed to the on-line testability of the microprocessors and the final actuation contacts for the ALWR.

Since the Evolutionary Requirements Document does not provide sufficient commitments, information, or design details regarding the above concerns, the staff will evaluate resolution of these concerns during its review of individual applications for FDA/DC.

See Section 3.2.39 for resolution of the open issue addressed in the staff's DSER for Chapter 5 (February 1990) for Issue 120.

Issue 120 has not yet been generically resolved by the staff and has a medium safety priority. Should the final agency resolution of this issue dictate additional actions for ALWR evolutionary plant designs, the staff will expect the FDA/DC applicant to address resolution of Issue 120 for staff evaluation.

3.2.46 121, Hydrogen Control for Large, Dry PWR Containments

Issue: As a result of the TMI-2 accident, the Commission promulgated regulatory requirements on hydrogen control in 10 CFR 50.34 and 50.44. 10 CFR 50.34(f) requires a hydrogen control system based on a 100-percent fuel-cladding metal-water reaction and a hydrogen concentration limit of 10 percent on uniformly distributed hydrogen in the containment or a postaccident atmosphere that will not support hydrogen combustion. Plants covered by this rule included only those whose construction permits had not been issued at the time of the TMI-2 accident. The issue is whether this requirement applies to the evolutionary ALWR.

EPRI Proposal: In Section 3.5.3 of Appendix B to Chapter 1 of the Evolutionary Requirements Document, EPRI originally proposed that the evolutionary plant design have the capability to ensure that necessary accident prevention and mitigation functions can be performed during and after events in which hydrogen is produced. The Evolutionary Requirements Document originally required that the design accommodate an amount of hydrogen equivalent to that generated by oxidation of 75 percent of the fuel cladding surrounding the active fuel. This was to be accomplished for a PWR dry containment by containment volume and mixing so that the uniformly distributed hydrogen gas concentration in the containment does not exceed 13 percent under dry conditions. For a BWR, the Evolutionary Requirements Document permitted provision of a noncombustible containment atmosphere as an acceptable alternative approach.

Staff Evaluation: In Appendix B of its DSER for Chapter 5, the staff concluded that EPRI's proposed approach to resolving this issue was unacceptable. The staff agreed with EPRI's proposal on hydrogen control for BWRs, indicating that inerting was an acceptable approach and would meet the requirements of 10 CFR 50.44.

In SECY-90-016, the staff recommended to the Commission that the hydrogen control requirements for evolutionary plants be identical to those stated in 10 CFR 50.34(f). This regulation specifically requires a hydrogen control

system that can safely accommodate an amount of hydrogen equivalent to that generated by the reaction of 100 percent of the fuel-cladding metal and that can ensure that uniformly distributed hydrogen gas concentrations in the containment do not exceed 10 percent by volume. The Commission approved the staff's recommendation in a staff requirements memorandum dated June 26, 1990.

In a letter dated December 6, 1991, EPRI stated that it intends to fully comply with the staff's position. Revision 4 to the Evolutionary Requirements Document revised Section 3.5.3. of Appendix B to Chapter 1 of the Evolutionary Requirements Document to require that ALWR designs be capable of accommodating 100-percent oxidation of active fuel cladding and either maintaining the uniformly distributed maximum containment hydrogen gas concentration below 10 percent or ensuring that the atmosphere is incapable of supporting hydrogen combustion. This is acceptable and resolves the open issue in the DSER. However, since the staff position in SECY-90-016 is that the plant-specific design must comply with 10 CFR 50.34(f) for combustible gas control, the staff will review relevant design features for conformance to 10 CFR 50.34(f) during its review of applications for FDA/DC. See also the evaluations in Sections 2.3 and 6.5.1 of Chapter 5 of this report and the evaluation of Issue A-48 in Section 3.2.10 of this appendix.

3.2.47 122.1.a, Davis-Besse Loss-of-All-Feedwater Event of June 9, 1985: Short-Term Actions - Failure of Isolation Valves in Closed Position

Issue 122.1.a was subsumed by Issue 124. See Section 3.2.52.

3.2.48 122.1.b, Davis-Besse Loss-of-All-Feedwater Event of June 9, 1985: Short-Term Actions - Recovery of Auxiliary Feedwater

Issue 122.1.b was subsumed by Issue 124. See Section 3.2.52.

3.2.49 122.1.c, Davis-Besse Loss-of-All-Feedwater Event of June 9, 1985: Short-Term Actions - Interruption of Auxiliary Feedwater Flow

Issue 122.1.c was subsumed by Issue 124. See Section 3.2.52.

3.2.50 122.2, Davis-Besse Loss-of-All-Feedwater Event of June 9, 1985: Short-Term Actions - Initiating Feed-and-Bleed

Issue: During the loss-of-feedwater event at Davis-Besse, the operators were reluctant to initiate feed-and-bleed cooling. The operators' actions raised concerns regarding the adequacy of emergency procedures, operator training, and available plant monitoring systems for determining the need to initiate feed-and-bleed cooling following the loss of the steam generator heat sink.

During the event at Davis-Besse, the steam generators became dry, meeting the criteria for the initiation of feed-and-bleed cooling. However, the operators did not initiate feed-and-bleed operations. The staff is concerned that procedures and/or training may not be adequate to ensure that operators initiate feed-and-bleed cooling when it is necessary to avert a core melt.

EPRI Proposal: Section 3 of Chapter 10 of the Evolutionary Requirements Document requires that the function and task analysis be used in the development of control room instrumentation and operating procedures and that it explicitly consider the actions of operators to control the plant.

Staff Evaluation: In its DSER for Chapter 10, the staff concluded that the design criteria in the Evolutionary Requirements Document were acceptable. However, it also concluded that since the Evolutionary Requirements Document does not provide the necessary details, the staff will evaluate the operator training program and emergency operating procedures during its review of individual applications for FDA/DC.

3.2.51 123, Deficiencies in the Regulations Governing Design-Basis-Accident and Single-Failure Criteria Suggested by the Davis-Besse Event of June 9, 1985

Issue 123 was mentioned in Appendix A of the staff's DSER for Chapter 3 (May 1988), but was not evaluated. In March 1992, Issue 123 was dropped as a generic issue; therefore, no evaluation is necessary.

3.2.52 124, Auxiliary Feedwater System Reliability

Issue: A function of the auxiliary feedwater (AFW) system in the majority of current plants is to supply water to the secondary side of the steam generators during system fill, normal plant heatup, normal plant hot standby, and normal plant cold shutdown. The AFW system also functions following loss of normal feedwater flow, including loss due to offsite power supply failure, and provides emergency feedwater (EFW) following such postulated accidents as a main feedwater line break or a main steamline break.

The loss of all feedwater at Davis-Besse resulted in an NRC investigation of the event that indicated that the potential inability to remove decay heat from the reactor core was due to the questionable reliability of the EFW system caused by any or all of the following:

- loss of all AFW as a result of the common-mode failure of the AFW pump discharge isolation valves to open
- excessive delay in recovering AFW because of difficulty in restarting AFW pump steam-driven turbines once they tripped
- interruption of AFW flow because of failures of features used to mitigate the effects of steamline and feedwater line break accidents

In addition, the investigation of the event indicated that (1) a two-train system with a steam turbine-driven AFW pump may not be able to achieve the desired level of reliability and (2) the provision to automatically isolate AFW from a steam generator affected by a main steam or feedwater line break may tend to increase the risk that adequate decay heat removal is not available rather than to decrease it.

EPRI Proposal: Section 5.3.1.2 of Chapter 5 of the Evolutionary Requirements Document states that the EFW system will be a dedicated safety-related system that will have no function for normal operation. The Evolutionary Requirements Document requires that

- A safety-grade supply of feedwater be provided of sufficient volume to permit safe cold shutdown, based on (1) a main feedwater line break without isolation of EFW flow to the affected steam generator for 30 minutes, (2) refill of the intact steam generators, (3) 8 hours of

operation at hot standby conditions, (4) subsequent cooldown of the reactor coolant system within 6 hours to conditions that permit operation of the residual heat removal system, and (5) continuous operation of one reactor coolant pump. (Section 5.3.2.3.1, Chapter 5)

- The EFW system use two noncondensing steam turbine-driven pumps and two electric motor-driven pumps. Any two pumps for four-steam-generator plants and any one pump for two-steam-generator plants must be capable of satisfying the flow requirements for licensing design-basis-accident conditions. Any single pump must be capable of satisfying the flow requirement for best-estimate evaluations of core damage frequency. (Section 5.3.3.1.2, Chapter 5)

- Each turbine-driven pump be supplied with steam only from the steam generator it supplies with feedwater. (Section 5.3.3.1.5.1, Chapter 5)

- A cavitating venturi be provided in the discharge line to each steam generator to prevent pump cavitation due to runout and also to minimize other potentially adverse effects of excessive EFW flow. (Section 5.3.3.1.8, Chapter 5)

- The rate of opening of the steam supply valves to the turbine-driven pumps be limited to the extent required to ensure reliable startup of the pumps. (Section 5.3.3.2.3, Chapter 5)

- Restart of the turbine-driven pumps following an overspeed trip be facilitated by ensuring ready access to pumps, by labeling the components required to reset the overspeed trip, and by requiring the manufacturer to provide a clear set of reset instructions to be posted adjacent to each pump. For designs that use an electronic trip that is set below the mechanical overspeed trip, reset capability from the control room must be provided. (Section 5.3.3.2.4, Chapter 5)

- Isolation and flow-regulating valves in the turbine-driven pump discharge lines be capable of performing their safety function independent of normal offsite and emergency onsite ac power. Manual capability must be provided to permit positioning of these valves in the event of a loss of power. (Section 5.3.3.4.2, Chapter 5)

- Automatic and manual initiation of EFW flow be provided. The EFW control system must have a safety-grade control system that will feed the steam generator at a maximum rate when the steam generator water level falls below the minimum steam generator water level assumed in the plant safety analysis. (Section 4.2.3.4, Chapter 3)

- The steam generator system be designed so that actuation of the EFW system is not required for at least 20 minutes following the point at which the low-level setpoint is reached. (Section 4.2.8.1, Chapter 3)

- The minimum flow delivered to the steam generators under licensing design-basis-accident conditions ensure adequate heat removal from the reactor coolant system. (Section 5.3.2.1.1, Chapter 5)

- The minimum flow delivered to the steam generators ensure adequate heat removal from the reactor coolant system for best-estimate evaluations of core damage frequency. (Section 5.3.2.1.3, Chapter 5)

- The plant designer define the mass and energy input to the containment resulting from flow of EFW to the affected steam generator following a main steamline break and ensure that this is accounted for in the containment design. Operator action to terminate EFW flow to the affected steam generator must not be assumed before 30 minutes. The system should not rely on automatic isolation of EFW to prevent containment overpressurization. (Section 5.3.2.2, Chapter 5)

- The plant designer perform an analysis of each automatic control loop of the EFW system to demonstrate capability for stable operation over the full range of operating conditions. (Section 5.3.3.5.1, Chapter 5.)

Staff Evaluation: In its DSER for Chapter 5 (February 1990), the staff concluded that EPRI's proposed resolution satisfactorily addressed the staff's AFW system reliability concerns. However, the staff will verify the acceptability of the analyses (discussed above) performed by the designer or applicant during its review of individual applications for FDA/DC.

3.2.53 125.II.7, Davis-Besse Loss-of-All-Feedwater Event of June 9, 1985: Long-Term Actions - Reevaluate Provision To Automatically Isolate Feedwater From Steam Generator During a Line Break

Automatic isolation of AFW from a steam generator is provided to mitigate the consequences of a steam or feedwater line break; however, it is needed only in a relatively rare event (steam or feedwater line break). In contrast, this isolation has many disadvantages. The safety significance of this issue arises from the fact that the negative aspects of automatic isolation involve accident sequences that have more frequent initiators and more significant consequences than those of the positive aspects. The staff concluded that removal of the AFW automatic isolation feature will neither result in a substantial safety improvement nor be cost effective and resolved the issue deciding to take no action. This does not preclude, however, an applicant from proposing the removal of the automatic isolation feature on the basis of plant-specific considerations.

See the staff's evaluation of Issue 124 in Section 3.2.52.

3.2.54 125.II.11, Davis-Besse Loss-of-All Feedwater Event of June 9, 1985: Long-Term Actions - Recovery of Main Feedwater as Alternative to Auxiliary Feedwater

This issue addressed alternative means of recovering feedwater should the AFW systems of PWRs fail. In resolving Issue 124, the staff evaluated the potential alternative recovery methods for both main and AFW systems for plants with two-train AFW systems. Contingent on implementation of the staff's recommendations proposed as the resolution of Issue 124, Issue 125.II.11 was dropped as a new and separate generic safety issue. See the staff's evaluation of Issue 124 in Section 3.2.52.

3.2.55 127, Maintenance and Testing of Manual Valves in Safety-Related Systems

Issue: Following a loss of power for the integrated control system at Rancho Seco, the operators attempted to close the auxiliary feedwater manual isolation valve. The isolation valve could not be moved, even with a valve wrench. The valve was seized because of inadequate lubrication as a result of a lack of preventive maintenance over the 10- to 12-year operational life of the plant. The staff informed licensees of this incident in IE Information Notice 86-61, "Failure of Auxiliary Feedwater Manual Isolation Valve." Because the estimated reduction in public risk would be minimal as a result of the resolution of this issue, Issue 127 was assigned a low safety priority.

EPRI Proposal: Section 5.6.3 of Chapter 10 of the Evolutionary Requirements Document states that the ALWR is expected to use a large amount of multiplexed data. This use will aid in plant operations, including maintenance planning and testing.

Staff Evaluation: In the DSER for Chapter 10, the staff agreed with EPRI that the implementation of the design criteria in the Evolutionary Requirements Document would resolve this issue. However, since a final resolution for this issue had not yet been developed, the staff concluded that any additional criteria resulting from the resolution of Issue 127 would have to be met by the ALWR designers.

Issue 127 has not yet been generically resolved by the staff and has a low safety priority. Unless the safety priority of this issue changes to medium or high, the FDA/DC applicant will not be expected to address the resolution of Issue 127.

3.2.56 128, Electrical Power Reliability

Issue: Concerns have been raised regarding the dependence on Class 1E power, especially dc power, of the decay heat removal systems required for long-term heat removal. Failure of one division would generally result in a reactor scram, which would then require removal of decay heat. The frequency of reported failures of single dc divisions gives rise to the concern that the second dc division may not be available.

Two of the specific reasons for the concern that safety-related power may be unreliable are addressed by Issue 128. One reason is that some operating nuclear power plants do not have technical specifications or administrative controls governing operational restrictions for Class 1E 120-V ac vital instrument buses and associated inverters. Without such restrictions, these power sources could be out of service indefinitely and might place certain safety systems in a situation where they could not meet the single-failure criterion. This is of particular concern during the period before the start and load of emergency diesel generators following a loss of offsite power. The other reason is that the design of some plants does not provide interlocks to prevent the inadvertent closure of the single tie breaker between Class 1E buses of all voltages.

EPRI Proposal: Section 3.11.1.4.1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document states that the concerns raised by this issue can be resolved by avoiding the use of bus tie breakers that could compromise

division independence and by providing a reliable dc power supply, especially when the failure of one dc power system leads to a reactor scram. Section 3.11.1.4.2 lists the following design criteria to address this issue:

- Section 2.2.8 of Chapter 5 requires that each division of the engineered safety systems requiring electric power be provided with an emergency onsite source of ac and/or dc power.

- Section 2.2.9 of Chapter 5 requires that at least two separate and independent connections be provided to offsite power sources capable of starting and running all Class 1E loads required for safe shutdown.

- Section 2.3.1 of Chapter 5 requires that the specified functions of engineered safety systems be met by the use of redundant divisions.

- Section 2.3.2 of Chapter 5 requires that the divisions of engineered safety systems be totally independent and separated both mechanically and electrically, except for areas in which it is physically impractical or less safe.

- Section 3.3.1 of Chapter 5 requires the decay heat removal systems to be redundant and safety grade.

- Section 4.2.1 of Chapter 5 requires that BWR designs have three independent divisions for the core coolant inventory control (CCIC) and decay heat removal (DHR) systems.

- Section 4.2.6.1 of Chapter 5 requires that each division of the CCIC and DHR systems for BWR designs have its own independent emergency ac and dc power source.

- Section 5.1.2.1 of Chapter 5 requires that PWR designs have two independent divisions for the CCIC and DHR functions.

- Section 5.2.3.1.1 of Chapter 5 requires that the PWR designs have two independent divisions for the residual heat removal function.

- Section 2.3.6 of Chapter 11 requires that each division of engineered safety systems requiring electric power be provided with an independent emergency onsite source of ac and dc power.

- Section 3.2.3 of Chapter 11 requires that at least two separate and independent connections for offsite power sources capable of starting and running all Class 1E loads required for safe shutdown be included in the design.

- Sections 2.3.6, 2.3.7, and 2.3.10 of Chapter 11 require separation of electrical power systems to preclude interactions that could adversely affect the functioning of the dc power systems. Specifically, the use of bus ties between safety divisions is prohibited.

- Sections 2.3.9 and 2.3.11 of Chapter 11 require that non-safety-related loads be placed on power supplies that are completely separate from those on which safety-related loads are placed.

- Section 7.2.1 of Chapter 11 requires that the loss of any plant battery or dc bus concurrent with a single independent failure in any other system required for shutdown cooling not result in a total loss of reactor cooling capability.

- Section 7.3.2.4 of Chapter 11 requires that each reactor protection channel be normally powered from a dedicated Class 1E source that is normally independent of other dc sources.

In summary, the Evolutionary Requirements Document states that each division of the engineered safety systems will have an emergency onsite source of ac and dc power and at least two connections for offsite power, all of which will be separate and independent. Specifically, there will be three independent divisions of decay heat removal for the BWR design and two for the PWR design, each with its own emergency ac and dc power source.

Staff Evaluation: The design criteria EPRI proposes for the evolutionary design adequately address the design concerns of Issue 128. Additional related concerns regarding the failure of a dc bus causing an event and coincident failure of another dc bus are discussed further in Chapter 11 of this report.

However, some concerns of Issue 128 that involve operational aspects need to be addressed by implementation of appropriate procedures and controls (e.g., through technical specifications). Since the Evolutionary Requirements Document does not provide sufficient details regarding plant operations to fully resolve these concerns of Issue 128, the staff will evaluate these items (including operations, maintenance, and testing) during its review of individual applications for FDA/DC.

Issue 128 has not yet been generically resolved by the staff and has a high safety priority. Should the final resolution of this issue dictate additional actions for ALWR evolutionary plant designs, the staff will expect the FDA/DC applicant to address the resolution of Issue 128 for staff evaluation.

3.2.57 130, Essential Service Water Pump Failures at Multiplant Sites

Issue: This issue was identified as a result of the vulnerability of Byron Unit 1 to core-melt sequences with Unit 2 not operational. While Unit 2 was under construction, it was necessary to make a third service water pump available to Unit 1 via a crosstie with one of the two Unit 2 essential service water (ESW) pumps. This plant-specific issue raised concerns relative to multiplant units that have only two ESW pumps per plant but have crosstie capabilities. A limited survey of Westinghouse plants helped to identify the generic applicability of vulnerabilities of multiplant configurations with only two ESW pumps per plant. In the multiplant configurations identified (approximately 16 plants), all plants can share ESW pumps via a crosstie between plants. Efforts to resolve this issue are to include a survey of Babcock and Wilcox and Combustion Engineering plants to determine if similar multiplant configurations with two ESW pumps per plant and crosstie capabilities exist in these vendors' designs and a survey of single-unit plants to determine if similar ESW vulnerabilities exist.

EPRI Proposal: Proposed elements for resolving for this issue are included in Section 3.8.1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document.

Staff Evaluation: In the DSER for Chapter 8, the staff reported that Issue 130 raised concerns about multiplant units that have only two ESW pumps per plant with crosstie capabilities. Accordingly, the specific core-melt and radiological risks (consequences) determined by the evaluation of Issue 130 pertain only to the generic model multiplant configuration with two ESW pumps per plant. Sections 5.2.1 and 5.2.2 of Chapter 8 of the Evolutionary Requirements Document specify that the safety service water (SSW) system will have two pumps per division (with three independent divisions for the BWR and two independent divisions for the PWR). Section 3.8.1.4 of Appendix B to Chapter 1 states that a failure of a single pump will not prevent the SSW system from performing its intended safety-related function, and at multiplant sites, the reliability of the SSW system at the other unit or units will not be affected.

The staff indicated, however, that this design approach addressed only a portion of the reliability problems being considered under Issue 130. Both pumps are required in a given division to meet licensing-basis requirements, such as cold shutdown within 36 hours with a concurrent single failure, which defeats the redundant division. The Evolutionary Requirements Document also prohibits crossties between divisions and between units if more than one unit is placed at a site. Although elimination of the crosstie capability achieves simplicity and precludes certain operator errors, there are certain negative aspects, such as loss of flexibility during recovery actions.

The staff stated that EPRI should further examine the reliability aspects associated with the ALWR ESW systems and propose enhancements to the designs, if warranted. It suggested that EPRI consider providing two separate and independent intake structures, as well as incorporating a crosstie capability between plants with the attendant flexibility in recovery actions.

Regarding crosstie capability, EPRI responded that it had reviewed the SSW system configuration and its reliability and performed a probabilistic risk assessment for three different cooling water configurations (to compare relative reliabilities). From these analyses, EPRI concluded that crossties would be provided only if the plant designer demonstrated a compelling safety, operability, or availability need. Consequently, EPRI has decided not to include crossties to provide flexibility in recovery actions because the system design with crossties would increase in complexity.

Since a final resolution of Issue 130 has not yet been developed, the staff will evaluate the resolution of this issue during its review of individual applications for FDA/DC.

Issue 130 has a high safety priority.

3.2.58 132, Residual Heat Removal Pumps Inside Containment

Issue: Residual heat removal (RHR) pumps located inside the containment that have not been qualified for a harsh environment cannot be given credit in licensing analyses for providing long-term decay heat removal.

EPRI Proposal: Section 5.2.3.1.4 of Chapter 5 of the Evolutionary Requirements Document specifies that all RHR pumps should be located outside the containment.

Staff Evaluation: In its DSER for Chapter 5 (February 1990), the staff concluded that the requirement that all RHR pumps be located outside the containment was a satisfactory resolution of this issue. However, Issue 132 has not yet been prioritized by the staff and a generic resolution has not been developed. The final resolution may dictate additional actions for ALWR evolutionary plant designs. Therefore, the staff will evaluate the resolution of this issue, as necessary, during its review of individual applications for FDA/DC.

3.2.59 135, Steam Generator and Steamline Overfill

Issue: This issue was initiated in 1986 to integrate all current generic issue activities on steam generator and steamline integrity. These activities started as a result of several steam generator tube rupture (SGTR) events that occurred in operating PWRs. At least one of the events led to overfilling of the secondary side of the steam generator. This overfilling resulted in water in the steamline, which in turn caused safety valves designed for steam service to pass water and resulted in unanalyzed loads on the steam piping.

The 14 subissues of Issue 67 regarding staff actions that are integrated under this issue are as follows:

(1) Improved Accident Monitoring (Issue 67.3.3)
(2) Reactor Vessel Inventory Measurement (Issue 67.3.4)
(3) Reactor Coolant Pump Trip (Issue 67.4.1)
(4) Control Room Design Review (Issue 67.4.2)
(5) Emergency Operating Procedures (Issue 67.4.3)
(6) Organizational Responses (Issue 67.6.0)
(7) Reactor Coolant System Pressure Control (Issue 67.9.0)
(8) Steam Generator Overfill (Issue 67.3.1)
(9) Reassessment of Radiological Consequences (Issue 67.5.1)
(10) Reevaluation of SGTR Design Basis (Issue 67.5.2)
(11) Improved Eddy Current Tests (Issue 67.7.0)
(12) Supplemental Tube Inspections (Issue 67.10.0)
(13) Integrity of Steam Generator Tube Sleeves (Issue 67.2.1)
(14) Denting Criteria (Issue 67.8.0)

Studies concentrating on these 14 subissues have provided a better understanding of steam generator and secondary system integrity, including water hammer and its effect on secondary system components and branch lines as well as radiological consequences. The conclusion resulting from the studies made under this issue was that Subissues 1-8 are resolved, Subissues 9-12 are being pursued independently, and Subissues 13 and 14 are of little safety significance and have been designated as regulatory impact issues.

The basis for the resolution of Subissues 1-7 was that the concerns were addressed in the Three Mile Island Action Plan and implemented by multiplant action (MPA) letters. For Subissue 8, all aspects of steam generator overfill

(SGOF), except SGOF by SGTR, were considered under Issue A-47 (former unresolved safety issue). The staff investigated the risk associated with SGOF by SGTR and found it to be a small contributor to core damage frequency. Therefore, this subissue was closed without any new requirements.

EPRI Proposal: Section 3.3.5.5 of Appendix B to Chapter 1 of the Evolutionary Requirements Document (Revision 1) states that "total resolution of this issue is not applicable to design and thus is not applicable to the ALWR Utility Requirements Document." However, the Evolutionary Requirements Document includes requirements pertaining to the following:

- clear access in the vicinity of the primary side manways (Section 4.4.1.4, Chapter 3)
- space and arrangements inside the steam generator primary channel head to facilitate inspection and repair (Section 4.6.2, Chapter 3)
- water level indicator of reactor vessel inventory during normal operation and postaccident conditions (Section 4.5, Chapter 3)
- automatic control of steam generator water level (Section 4.2.3.2.1, Chapter 3)
- design to prevent secondary-side safety valve actuation for an SGTR (Section 4.2.5, Chapter 3)
- steam generator materials (Section 4.4.1.1, Chapter 3)
- tube bundle arrangement (Section 4.4.1.2.1, Chapter 3)
- prevention of water hammer in the steam generators (Section 4.4.1.8, Chapter 3)
- minimizing emergency feedwater flow (Section 5.3.3.1.8, Chapter 5)
- a safety depressurization and vent system to depressurize the reactor coolant system during an SGTR event (Section 5.5, Chapter 5)

The Evolutionary Requirements Document also states that "this issue has several elements, many of which are not yet defined or even assessed.... [I]t is not clear that the requirements directly address this issue."

Staff Evaluation: The staff agrees with the inclusion of the requirements listed above and the statement in the Evolutionary Requirements Document regarding the not yet defined and assessed elements of this issue. However, it is not clear why EPRI states that total resolution of this issue is not applicable to the evolutionary design and what is EPRI's position regarding the actions required by the MPA letters. Consequently, the staff will evaluate the resolution of Issue 135 during its review of individual applications for FDA/DC using the appropriate requirements and guidance current at that time.

3.2.60 142, Leakage Through Electrical Isolators in Instrumentation Circuits

Issue: This issue addresses electrical isolators used to maintain electrical separation between safety-related and non-safety-related electrical systems in nuclear power plants, thus preventing malfunctions in the non-safety-related systems from degrading the performance of safety-related circuits. The primary concern is that the amount of energy that could pass through certain types of isolation devices (and be transmitted to safety-related circuitry) during certain electrical transients might damage or seriously degrade the performance of Class 1E components or cause the isolation devices to give a false output, or the electrically generated noise on the circuit might cause the isolation device to give a false output.

EPRI Proposal: Not provided in Section 3 of Appendix B to Chapter 1 of the Evolutionary Requirements Document.

Staff Evaluation: Section 3.1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document states that it specifically addresses compliance with high- and medium-priority and nearly resolved generic safety issues included as of January 1990 in NRC's generic issue management control system (GIMCS). As of December 31, 1989, although Issue 142 was identified in Appendix B of NUREG-0933 as being applicable to future plants, it had not yet been prioritized.

As of the first-quarter fiscal year 1992 update of the GIMCS report, Issue 142 had a medium safety priority. Since this issue is not addressed in the Evolutionary Requirements Document, the staff will evaluate its resolution during its review of individual applications for FDA/DC.

3.2.61 143, Availability of Chilled Water Systems and Room Cooling

Issue: In recent years, several nuclear power plants have experienced problems with safety system components and control systems that were caused by a partial or total loss of heating, ventilating, and air conditioning (HVAC) systems. Many of these problems exist because of the desire to provide increased fire protection and the need to avoid severe temperature changes in equipment control circuits. Since the Browns Ferry fire, considerable effort has been expended to improve the fire protection of equipment required for safe shutdown. Generally, this improvement has been made by enclosing the affected equipment in small, isolated rooms. The result has been a significant increase in the impact of the loss of room cooling. Plant control and safety have improved with the introduction of electronic integrated circuits; however, these circuits are more susceptible to damage from severe changes in temperature caused by the loss of room cooling.

It is believed that failures of air cooling systems for areas housing key components, such as residual heat removal pumps, switchgear, and diesel generators, could contribute significantly to core-melt probability in certain plants. Because corrective measures are often taken at the affected plants once such failures occur, the impact of these failures on the proper functioning of air cooling systems has not been considered and plants with similar, inherent deficiencies may not be aware of these problems.

EPRI Proposal: Not provided in Section 3 of Appendix B to Chapter 1 of the Evolutionary Requirements Document.

Staff Evaluation: Section 3.1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document states that it specifically addresses compliance with high- and medium-priority and nearly resolved generic safety issues included as of January 1990 in NRC's GIMCS. As of December 31, 1989, although Issue 143 was identified in Appendix B of NUREG-0933 as being applicable to future plants, it had not yet been prioritized.

As of the first-quarter fiscal year 1992 update of the GIMCS report, Issue 143 had a high safety priority. Since this issue is not addressed in the Evolutionary Requirements Document, the staff will evaluate its resolution during its review of individual applications for FDA/DC.

3.2.62 151, Reliability of Recirculation Pump Trip During an Anticipated Transient Without Scram

Issue: Issue summary was not published in NUREG-0933 when this appendix was prepared.

EPRI Proposal: Not provided in Section 3 of Appendix B to Chapter 1 of the Evolutionary Requirements Document.

Staff Evaluation: Section 3.1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document states that it specifically addresses compliance with high- and medium-priority and nearly resolved generic safety issues included as of January 1990 in NRC's GIMCS. As of December 31, 1989, although Issue 151 was identified in Appendix B of NUREG-0933 as being applicable to future plants, it had not yet been prioritized.

As of the first-quarter fiscal year 1992 update of the GIMCS report, Issue 151 had a medium safety priority. Since this issue is not addressed in the Evolutionary Requirements Document, the staff will evaluate its resolution during its review of individual applications for FDA/DC.

3.2.63 153, Loss of Essential Service Water in Light-Water Reactors

Issue: This issue addresses the potential unavailability of the essential service water (ESW) system for all LWRs except those seven multiplant sites addressed under Issue 130. The ESW system at a nuclear power plant supplies cooling water to transfer heat from various safety-related and non-safety-related systems and equipment to the ultimate heat sink of the plant. It is known by different names at various types of plants. The design and operational characteristics of the ESW system are different for PWRs and BWRs. In addition, these characteristics may differ significantly in each of these reactor types.

Under Issue 153, the staff will examine all potential causes for ESW system unavailability, except those that are considered to be resolved by implementing the resolutions addressed in GL 89-13, such as biofouling, sediment, corrosion, and erosion (Issue 51). The safety concerns of this issue include partial or complete loss of ESW system functions resulting from common causes (such as icing of the intake structure), degradation of the ESW system, design deficiencies, and procedural or maintenance errors. A complete loss of the ESW system could lead to a core-melt accident, posing a significant risk to the public.

EPRI Proposal: Not provided in Section 3 of Appendix B to Chapter 1 of the Evolutionary Requirements Document.

The design features for the safety service water (SSW) system provided in Section 5.2 of Chapter 8 of the Evolutionary Requirements Document are summarized as follows:

- Performance Requirements
  - The SSW system will be designed to meet the required heat loads.
  - The SSW system will be provided with two pumps and two heat exchangers per division.
  - The plant designer will provide analyses for all potential operating conditions that properly account for uncertainties.
- System Arrangement
  - The SSW system will be divided into approximately equal-sized divisions, two for the PWR and three for the BWR.
  - A division will be made up of independent piping systems, each with pumps, heat exchangers, strainers, controls and instrumentation, power supplies, and associated equipment required for regulating system flow.

Staff Evaluation: Section 3.1 of Appendix B to Chapter 1 of the Evolutionary Requirements Document states that it specifically addresses compliance with high- and medium-priority and nearly resolved generic safety issues included as of January 1990 in NRC's GIMCS. Issue 153 was identified after December 31, 1989, and, therefore, was not included in Appendix B to NUREG-0933 until the June 30, 1990, update as being applicable to future plants; it was assigned a medium safety priority in February 1991.

The performance requirements and system arrangement for the SSW system indicated above do not adequately address the safety concerns of Issue 153. These concerns include partial or complete loss of ESW system functions resulting from common causes, degradation of the SSW system, design deficiencies, and procedural or maintenance errors. The Evolutionary Requirements Document should provide an assessment of these potential failure modes and their associated contributions to the core damage frequency and should identify dominant accident sequences. The Evolutionary Requirements Document states that the plant designer will be responsible for addressing these concerns for the future plant-specific design.

Since the Evolutionary Requirements Document does not provide sufficient information, the staff will evaluate resolution of Issue 153 during its review of individual applications for FDA/DC.

3.2.64 HF 4.4, Procedures - Guidelines for Upgrading Other Procedures

Issue: This issue addresses whether the staff needs to develop technical guidance for industry use in upgrading both normal and abnormal operating

procedures. Future work in this area includes the development of guidelines for the review of programs to upgrade procedures at nuclear power plants.

EPRI Proposal: In the note to Table B.3-1 in Appendix B to Chapter 1 of the Evolutionary Requirements Document, EPRI states that Issue HF 4.4 was determined, through the NUREG-1197 screening process, to be not technically relevant to the ALWR design. Nevertheless, the Evolutionary Requirements Document addresses this issue in Section 3.10.3 of Appendix B to Chapter 1.

Staff Evaluation: Appendix A to NUREG-1197 identifies issues that the staff determined were not applicable to the ALWR Program according to six categories. Issue HF 4.4 is listed as a Code (or Category) 5 item, that is, an issue not applicable to plant design, such as plant operation and operating procedures, management of operations, accident management and emergency preparedness, operator training and qualifications, inspection and maintenance, and operating experience reporting. By using the definition of Appendix A to NUREG-1197 for nonapplicability of generic safety issues to the ALWR Program, Issue HF 4.4 would be categorized as not applicable because it addresses procedures.

Despite this information, the Evolutionary Requirements Document addresses Issue HF 4.4 (in apparent conflict with the note to Table B.3-1 in Appendix B to Chapter 1 discussed above) stating in Section 3.10.3.4.3 of Appendix B to Chapter 1 that plant operating procedures will be an integral part of the plant design. As a result, the staff has evaluated the proposed elements for resolving this issue.

In its DSER for Chapter 10, the staff concluded that the information in the Evolutionary Requirements Document was not sufficient to determine if the requirements for the design, development, and validation of plant procedures were acceptable. The staff stated that although the electronic display of procedures might enhance information display flexibility, the limitations and constraints associated with this technology, as well as the operability, maintainability, and reliability of this technology, should be fully evaluated in the context of the entire control room and other control station designs before committing to such an approach. EPRI also should evaluate issues concerning the use of mixed types of procedures from one control station to the next, as well as the requirement for the active simulator to be used for the validation of procedures. Since the Evolutionary Requirements Document still does not provide sufficient information regarding these concerns, the staff will evaluate the resolution of Issue HF 4.4 during its review of individual applications for FDA/DC.

Issue HF 4.4 has not yet been generically resolved by the staff and has a high safety priority. Should the final agency resolution of this issue dictate additional actions for ALWR evolutionary plant designs, the staff will expect the FDA/DC applicant to address the resolution of Issue HF 4.4 for staff evaluation.

3.2.65 HF 5.1, Man-Machine Interface - Local Control Stations, and

3.2.66 HF 5.2, Man-Machine Interface - Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation

Issue: To ensure that the man-machine interface (M-MI) is adequate for the safe operation and maintenance of nuclear power plants, the staff concluded that the following should be developed: (1) human factors engineering guidelines for correcting M-MI problems and (2) regulatory guidance for integrating human factors engineering into new designs and into advanced technological improvements incorporated into existing designs.

Issue HF 5.1 addresses the expansion of efforts to provide guidance for the development of local control stations and auxiliary operator interfaces. Regulatory efforts on M-MIS have been limited to the control room and the remote shutdown panel. Additional guidance on improvements of existing annunciator systems may also result from these efforts. The staff expects to perform job and task analyses of control room crew activities to identify and describe communication and control links between the control room and the auxiliary control stations. In addition, the staff will analyze the functions of the auxiliary personnel from the task analyses to estimate the potential impact of errors by auxiliary personnel on the safety of the plant.

Issue HF 5.2 addresses the development of guidelines on the use and evaluation of improved annunciator systems that use advanced technologies. The staff is concerned that the human engineering guidelines used in current nuclear power plant control rooms, although adequate for current designs, may not be sufficient for the advanced and developing technologies that will be introduced into existing and future designs.

Solutions for these issues are expected to result in modifications of regulatory guides, the Standard Review Plan, and other guidance.

EPRI Proposal: In Section 3.10.3 of Appendix B to Chapter 1 of the Evolutionary Requirements Document, EPRI states that new and improved designs and systems, such as annunciator systems, computer systems, and other operational aids, will be required in a design that meets the design criteria of the Evolutionary Requirements Document to improve the M-MI. EPRI also states that a number of design features are required in the Evolutionary Requirements Document that address these human factors issues, including the following:

- Section 8.2.1.2 of Chapter 1 requires that human factors design principles be consistently applied throughout the design process for each operation workspace to reduce operation errors during all plant modes.

- Section 11.1.3.1 of Chapter 1 requires that the design process emphasize the human-machine interfaces and that an ongoing analysis be conducted to ensure that these requirements will be met.

- Section 11.10.4 of Chapter 1 requires that the designer provide a plant simulator/performance model that can be used to study the human engineering aspects of the plant controls and control room design.

- Section 11.11 of Chapter 1 requires that the interdisciplinary design review group include one member knowledgeable about the principles of human factors.

- Section 2.1.1 of Chapter 10 requires that operator limitations not be challenged.

- Section 2.2.10 of Chapter 10 requires the application of advanced technology to the main control room to enhance the design.

- Section 4 of Chapter 10 requires the application of human factors engineering principles to all control stations.

- Section 4.3 of Chapter 10 requires annunciator alarms to be addressed.

- Sections 3.4.5 and 5.6 of Chapter 10 require operator aids to be addressed.

- Section 3 of Chapter 10 requires the specific identification of functions and tasks and their allocation among the operators and automatic systems.

- Section 4.1.5.2 of Chapter 10 requires that the M-MI system designer develop and verify human factors practices for advanced M-MI technology where there is limited published guidance.

EPRI considers that the implementation of these requirements will satisfy NRC concerns about local control stations and the use of new, advanced technology in the design of ALWRs.

Staff Evaluation:

(1) Local Control Stations

In the DSER for Chapter 10, the staff determined that the commitment in the Evolutionary Requirements Document to design local control stations (LCSs) to the same standards as control stations in the main control room was acceptable. However, it concluded that EPRI should

- Provide a clear definition of LCSs in the Evolutionary Requirements Document. Issue HF 5.1 currently includes single-component interfaces, such as manually operated valves, in its definition.

- Address the issue of functional centralization in the Evolutionary Requirements Document, since a key element in the handling of safety functions is the degree of centralization that exists, which is reflected in the number of different LCSs required to carry out safety functions. As the number of LCSs increases, so does the time required to execute procedures and the workload associated with crew coordination, communication, and verification of operating procedures. Each introduces potential sources of human error that can be expected to increase with the number of panels involved.

Since the Evolutionary Requirements Document still does not provide sufficient information regarding these concerns, the staff will evaluate the resolution of Issue HF 5.1 during its review of individual applications for FDA/DC.

(2) Annunciators

In the DSER for Chapter 10, the staff determined that the Evolutionary Requirements Document integrates human factors engineering for the alarm system into the design of the hardware and software and that it requires that

certain human factors tests and evaluations of the alarm system be part of the overall test and evaluation program. However, EPRI should require additional human factors test and evaluation activities, including the development of

- a human factors verification and validation test plan
- a method to document test activities to provide traceability and ensure that all human factors requirements are addressed during tests and evaluations
- quantitative measures to assess human-system performance

Since the Evolutionary Requirements Document still does not provide sufficient information regarding these concerns, the staff will evaluate the resolution of Issue HF 5.2 during its review of individual applications for FDA/DC.

Issues HF 5.1 and HF 5.2 have not yet been generically resolved by the staff and both have a high safety priority. Should the final resolution of these issues dictate additional actions for ALWR evolutionary plant designs, the staff will expect the FDA/DC applicant to address the resolution of Issues HF 5.1 and HF 5.2 for staff evaluation.

Table 3B.1 Generic Safety Issues Addressed in the DSERs

Item/ Issue	Title	DSER Chap- ter	DSER Status
II.E.4.3	Containment Design - Integrity Check	5	**Open issue** (improved capability to detect inadvertent containment bypass)
A-29	Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage	9	**Open issue** (insider sabotage)
A-30	Adequacy of Safety-Related DC Power Supplies	11	Integrated into Issue 128
A-44	Station Blackout [former unresolved safety issue (USI)]	5	Resolved for Evolutionary Requirements Document
A-45	Shutdown Decay Heat Removal Requirements (former USI)	5	Resolved for Evolutionary Requirements Document
A-46	Seismic Qualification of Equipment in Operating Plants (former USI)	6	**Open issue** (emphasis on tank anchorage and installation in the Evolutionary Requirements Document)
A-47	Safety Implications of Control Systems (former USI)	10	**Confirmatory issue** (commitment to Generic Letter 89-19 in Evolutionary Requirements Document); **vendor item** (acceptability of design details); see also Issue 76
A-48	Hydrogen Control Measures and and Effects of Hydrogen Burns on Safety Equipment	5	**Open issue** (zirconium-water reaction and hydrogen generation limits and control); see also Issue 121
B-17	Criteria for Safety-Related Operator Actions	10	**Open issue** (justification for not complying with proposed revisions to ANSI/ANS 58.8)
B-22	LWR Fuel	4	Resolved for Evolutionary Requirements Document
B-29	Effectiveness of Ultimate Heat Sinks	8	Resolved for Evolutionary Requirements Document
B-32	Ice Effects on Safety-Related Water Supplies	8	Not evaluated; see also Issues 51 and 130

Table 3B.1 (Continued)

Item/ Issue	Title	DSER Chap- ter	DSER Status
C-8	Main Steamline Isolation Valve Leakage Control Systems	3	**Open issue** (pending final NRC resolution)
D-2	Emergency Core Cooling System Capability for Future Plants	5	Resolved for Evolutionary Requirements Document on the basis of commitment to satisfy the requirements of the Commission's Severe Accident Policy Statement
2	Failure of Protective Devices on Essential Equipment	10	**Open issue** (specific guidelines on exceptions to independence criteria for electrical systems); see also Issue 110
23	Reactor Coolant Pump Seal Failures	3, 8	**Open issue** (independent reactor coolant pump seal cooling during station blackout)
29	Bolting Degradation or Failure in Nuclear Power Plants	3	**Open issue**
51	Improving the Reliability of Open-Cycle Service Water Systems	8	**Open issue** (guidance in Generic Letter 89-13 and Information Notice 89-16); see also Issues B-32 and 130
67.7.0	Improved Eddy Current Tests	3	Resolved for Evolutionary Requirements Document
70	Power-Operated Relief Valve and Block Valve Reliability	5	**Open issue** (specification of criterion for depressurization rate of safety depressurization and vent system)
75	Generic Implications of ATWS (Anticipated Transient Without Scram) Events at the Salem Nuclear Plant	10	Resolved for Evolutionary Requirements Document except for: **Open issue** (justification of 14-day design objective for corrective maintenance of man-machine interface system equipment);

Table 3B.1 (Continued)

Item/ Issue	Title	DSER Chap- ter	DSER Status
			Vendor items (post-trip review program and procedures; equipment classification and vendor interface for reactor trip system components and all preventative maintenance and safety-related equipment; details of surveillance program for reactor trip breakers)
76	Instrumentation and Control Power Interactions	10	**Confirmatory issue** (commitment to Generic Letter 89-19 in Evolutionary Requirements Document); **Vendor item** (acceptability of design details); see also Issue A-47
79	Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown	4	Resolved for Evolutionary Requirements Document
82	Beyond-Design-Basis Accidents in Spent Fuel Pools	7	**Open issue** (low-density storage racks for most recently discharged fuel)
84	Combustion Engineering Power-Operated Relief Valves	5	**Open issue** (specification of criterion for depressurization rate of safety depressurization and vent system)
91	Main Crankshaft Failures in Transamerica Delaval Emergency Diesel Generators	11	Resolved for Evolutionary Requirements Document
93	Steam Binding of Auxiliary Feedwater Pumps	5	Resolved for Evolutionary Requirements Document
94	Additional Low-Temperature Overpressure Protection for Light-Water Reactors	3, 4	**Open issue** (pending final NRC resolution)
96	Residual Heat Removal Suction Valve Testing	5	**Open issue** (pending NRC's completion of intersystem loss-of-coolant accident

Table 3B.1 (Continued)

Item/ Issue	Title	DSER Chap-ter	DSER Status
			resolution program for BWRs and PWRs); see also Issues 99, 105, and 120
99	Reactor Coolant System/Residual Heat Removal Suction Line Valve Interlock on PWRs	5	**Open issue** (pending NRC's completion of intersystem loss-of-coolant accident resolution program for BWRs and PWRs); see also Issues 96, 105, and 120
101	BWR Water Level Redundancy	10	**Confirmatory issue** (commitment to comply with Generic Letter 89-11); **vendor item** (review of specific designs for water level instrumentation)
103	Design for Probable Maximum Precipitation	6, 8	**Open issue** (reference to SRP Section 2.4.10 and Generic Letter 89-22 in Evolutionary Requirements Document)
105	Interfacing Systems Loss-of-Coolant Accident at LWRs	5	**Open issue** (pending NRC's completion of intersystem loss-of-coolant accident resolution program for BWRs and PWRs); see also Issues 96, 99, and 120
107	Main Transformer Failures	11	**Open issue** (pending resolution of open issue on transformer location in Chapter 6)
110	Equipment Protective Devices on Engineered Safety Features	10	**Open issue** (specific guidelines on exceptions to independence criteria for electrical systems); see also Issue 2
113	Dynamic Qualification Testing of Large-Bore Hydraulic Snubbers	3	Resolved for Evolutionary Requirements Document
115	Enhancement of the Reliability of Westinghouse Solid State Protection System	10	Resolved for Evolutionary Requirements Document

Table 3B.1 (Continued)

Item/ Issue	Title	DSER Chap- ter	DSER Status
116	Accident Management for Future Plants	10	Issue subsumed into staff's efforts on accident management; to be handled under open issue in Chapter 5 on severe accident management program
117	Allowable Time for Diverse Simultaneous Equipment Outages	5	Resolved for Evolutionary Requirements Document
118	Tendon Anchorage Failure	6	Not applicable
120	On-Line Testability of Protection Systems	5	**Open issue** (pending NRC's completion of intersystem loss-of-coolant accident resolution program for BWRs and PWRs); see also Issues 96, 99, and 105
121	Hydrogen Control for Large, Dry PWR Containments	5	**Open issue** (zirconium-water reaction and hydrogen generation limits and control); see also Issue A-48
122.1.a	Davis-Besse Loss-of-All-Feedwater Event of June 9, 1985: Short-Term Actions - Failure of Isolation Valves in Closed Position	5	Resolved for Evolutionary Requirements Document; **Vendor item** (acceptability of designers' analyses); see also Issues 122.1.b, 122.1.c, 124, 125.II.7, and 125.II.11
122.1.b	Davis-Besse Loss-of-All-Feedwater Event of June 9, 1985: Short-Term Actions - Recovery of Auxiliary Feedwater	5	Resolved for Evolutionary Requirements Document; **Vendor item** (acceptability of designers' analyses); see also Issues 122.1.a, 122.1.c, 124, 125.II.7, and 125.II.11
122.1.c	Davis-Besse Loss-of-All-Feedwater Event of June 9, 1985: Short-Term Actions - Interruption of Auxiliary Feedwater Flow	5	Resolved for Evolutionary Requirements Document; **Vendor item** (acceptability of designer's analyses); see also Issues 122.1.a, 122.1.b, 124, 125.II.7, and 125.II.11

Table 3B.1 (Continued)

Item/ Issue	Title	DSER Chap- ter	DSER Status
122.2	Davis-Besse Loss-of-All-Feedwater Event of June 9, 1985: Short-Term Actions - Initiating Feed-and-Bleed	10	Resolved for Evolutionary Requirements Document; **Vendor item** (operator training program)
123	Deficiencies in the Regulations Governing Design-Basis-Accident and Single-Failure Criteria Suggested by the Davis-Besse Event of June 9, 1985	3	Reassigned to Chapter 1 of Evolutionary Requirements Document; to be addressed in the final safety evaluation report
124	Auxiliary Feedwater System Reliability	5	Resolved for Evolutionary Requirements Document; **Vendor item** (acceptability of designers' analyses); see also Issues 122.1.a, 122.1.b, 122.1.c, 125.II.7, and 125.II.11
125.I.3	Davis-Besse Loss-of-All-Feedwater Event of June 9, 1985: Long-Term Actions - Safety Parameter Display System Availability	10	Resolved for Evolutionary Requirements Document
125.I.4	Davis-Besse Loss-of-All-Feedwater Event of June 9, 1985: Long-Term Actions - Plant-Specific Simulator	10	Resolved for Evolutionary Requirements Document
125.I.5	Davis-Besse Loss-of-All-Feedwater Event of June 9, 1985: Long-Term Actions - Safety Systems Tested in All Conditions Required by Design-Basis Accident	10	Resolved for Evolutionary Requirements Document
125.I.6	Davis-Besse Loss-of-All-Feedwater Event of June 9, 1985: Long-Term Actions - Valve Torque Limit and Bypass Switch Settings	10	Dropped as a generic issue; no evaluation needed

Table 3B.1 (Continued)

Item/ Issue	Title	DSER Chap-ter	DSER Status
125.II.7	Davis-Besse Loss-of-All-Feedwater Event of June 9, 1985: Long-Term Actions - Reevaluate Provision To Automatically Isolate Feedwater From Steam Generator During a Line Break	5	Resolved for Evolutionary Requirements Document; **Vendor item** (acceptability of designers' analyses); see also Issues 122.1.a, 122.1.b, 122.1.c, 124, and 125.II.11
125.II.11	Davis-Besse Loss-of-All-Feedwater Event of June 9, 1985: Long-Term Actions - Recovery of Main Feedwater as Alternative to Auxiliary Feedwater	5	Resolved for Evolutionary Requirements Document; **Vendor item** (acceptability of designers' analyses); see also Issues 122.1.a, 122.1.b, 122.1.c, 124, and 125.II.7
126	Reliability of PWR Main Steam Safety Valves	3	Resolved for Evolutionary Requirements Document
127	Maintenance and Testing of Manual Valves in Safety-Related Systems	10	Resolved for the Evolutionary Requirements Document
128	Electrical Power Reliability	11	**Open issue** (listing of operational, maintenance, and testing issues of Issue 128 in Evolutionary Requirements Document)
130	Essential Service Water Pump Failures at Multiplant Sites	8	**Open issue** (needs to be addressed); see also Issues B-32 and 51
132	Residual Heat Removal Pumps Inside Containment	5	Resolved for Evolutionary Requirements Document
HF 4.4	Procedures - Guidelines for Upgrading Other Procedures	10	**Open issue** (electronic display of procedures and mixed types of procedures)
HF 5.1	Man-Machine Interface - Local Control Stations	10	**Open issue** (definition of local control stations and centralization); see also Issue HF 5.2
HF 5.2	Man-Machine Interface - Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation	10	**Open issue** (definition of local control stations and centralization); see also Issue HF 5.1

Table 3B.2 Safety Issues Applicable to Future Plants That Have Not Been Prioritized*

Issue	Title
2*	Failure of Protective Devices on Essential Equipment
76*	Instrumentation and Control Power Interactions
78	Monitoring of Fatigue Transient Limits for Reactor Coolant System
89	Stiff Pipe Clamps
110*	Equipment Protective Devices on Engineered Safety Features
118*	Tendon Anchorage Failure
132*	Residual Heat Removal Pumps Inside Containment
144	Scram Without a Turbine-Generator Trip
145	Improve Surveillance and Startup Testing Programs
146	Support Flexibility of Equipment and Components
147	Fire-Induced Alternate Shutdown Control Room Panel Interactions
148	Smoke Control and Manual Firefighting Effectiveness
149	Adequacy of Fire Barriers
152	Design Basis for Valves That Might Be Subjected to Significant Blowdown Loads
154	Adequacy of Emergency and Essential Lighting
155.1	Generic Concerns Arising From TMI-2 Cleanup - More Realistic Source Term Assumptions
155.2	Generic Concerns Arising From TMI-2 Cleanup - Establish Licensing Requirements for Nonoperating Facilities
155.3	Generic Concerns Arising From TMI-2 Cleanup - Improve Design Requirements for Nuclear Facilities
155.4	Generic Concerns Arising From TMI-2 Cleanup - Improve Criticality Calculations
155.5	Generic Concerns Arising From TMI-2 Cleanup - More Realistic Severe Reactor Accident Scenario
155.6	Generic Concerns Arising From TMI-2 Cleanup - Improve Decontamination Regulations
155.7	Generic Concerns Arising From TMI-2 Cleanup - Improve Decommissioning Regulations
156.1.1	Systematic Evaluation Program Issues - Settlement of Foundations and Buried Equipment

*Although not prioritized, the evolutionary DSER chapters addressed Issues 2, 76, 110, 118, and 132. See Table 3B.1.

Table 3B.2 (Continued)

Issue	Title
156.1.2	Systematic Evaluation Program Issues - Dam Integrity and Site Flooding
156.1.3	Systematic Evaluation Program Issues - Site Hydrology and Ability To Withstand Floods
156.1.4	Systematic Evaluation Program Issues - Industrial Hazards
156.1.5	Systematic Evaluation Program Issues - Tornado Missile
156.2.1	Systematic Evaluation Program Issues - Severe Weather Effects on Structures
156.2.2	Systematic Evaluation Program Issues - Design Codes, Criteria, and Load Combinations
156.2.3	Systematic Evaluation Program Issues - Containment Design and Inspection
156.2.4	Systematic Evaluation Program Issues - Seismic Design of Structures, Systems, and Components
156.3.1.1	Systematic Evaluation Program Issues - Shutdown Systems
156.3.1.2	Systematic Evaluation Program Issues - Electrical Instrumentation and Controls
156.3.2	Systematic Evaluation Program Issues - Service and Cooling Water Systems
156.3.3	Systematic Evaluation Program Issues - Ventilation Systems
156.3.6.1	Systematic Evaluation Program Issues - Emergency AC Power
156.3.6.2	Systematic Evaluation Program Issues - Emergency DC Power
156.3.8	Systematic Evaluation Program Issues - Shared Systems
156.4.2	Systematic Evaluation Program Issues - Testing of the Reactor Protection System and Engineered Safety Features
156.6.1	Systematic Evaluation Program Issues - Pipe Break Effects on Systems and Components

Table 3B.3 Newly Identified Safety Issues

Issue	Title
157	Containment Performance Improvement Program
158	Performance of Power-Operated Valves Under Design-Basis Conditions
159	Qualification of Safety-Related Pumps While Running on Minimum Flow
160	Spurious Actions of Instrumentation Upon Restoration of Power
161	Use of Non-Safety-Related Power Supplies in Safety-Related Circuits
162	Inadequate Technical Specifications for Shared Systems at Multi-plant Sites When One Unit Is Shut Down

4 REGULATORY DEPARTURE ANALYSES

In the staff requirements memorandum (SRM) dated August 24, 1989, the Commission instructed the staff to provide an analysis detailing where the staff proposes departure from current regulations or where it is substantially supplementing or revising interpretive guidance applied to currently licensed LWRs. The staff considers these to be policy issues. In this section, the staff discusses the regulatory departure analyses of those issues identified for the evolutionary plant designs.

During its review of the EPRI Requirements Document, the staff identified a number of issues significant to reactor safety as it considered operating experience, probabilistic risk assessment studies, and evaluations of the evolutionary and passive ALWR designs. In Commission papers SECY-90-016 and SECY-91-078, the staff proposed resolutions for some of these issues for evolutionary designs. In addition, the staff developed draft Commission papers, "Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements," and "Design Certification and Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light Water Reactor Designs," that were issued on February 27 and July 6, 1992, respectively. These draft Commission papers address the status of those issues identified previously as well as new issues that pertain to both the evolutionary and passive LWR designs. The staff considers the discussions in these papers to be its regulatory departure analyses of the policy issues identified to date for the evolutionary plant designs.

The staff also forwarded other policy papers to the Commission that address the status of the staff's review of other issues related to the reviews of the ALWR. A list of these papers is provided in Appendix E to Volume 1 of this report.

Some of the issues discussed in these papers represent matters specific to the design of these facilities; others address the implementation of the design certification process of 10 CFR Part 52. In the Commission papers, the staff identified those instances in which its positions differ from current regulatory requirements or in which the staff is substantially supplementing or revising interpretive guidance applied to currently licensed LWRs. A discussion of the nature of the current regulatory requirements or their interpretation, the positions of the ALWR vendors and of EPRI, and, where available, the resolution that the staff is proposing, including the departure, if any, from current regulatory requirements and practice, and the basis for the staff's position are given for each issue. To aid in identifying its positions, the staff underlined those for which it requested Commission approval.

These issues are considered fundamental to agency decisions on the acceptability of the ALWR designs. For easy identification, Table 4B-1 lists the issues that are applicable to the Evolutionary Requirements Document with a cross-reference to the chapters and sections of this report in which they are discussed.

The Commission addressed SECY-90-016 and SECY-91-078 in its SRM of June 26, 1990, and August 15, 1991, respectively. The Advisory Committee on Reactor Safeguards (ACRS) forwarded its comments on these papers to the Commission by letters dated April 26, 1990, and April 23, 1991. The staff's responses to the comments of the ACRS were issued on April 27, 1990, and May 15, 1991.

The February 27 and July 6, 1992, draft Commission papers have been forwarded to the ACRS. The ACRS forwarded its comments on the February 27, 1992, draft policy paper by letters dated May 13 and August 17, 1992. The staff's response to the May 13, 1992, comments of the ACRS was issued on June 12, 1992.

The staff will include the views of the ACRS in the final papers and document its final positions before seeking Commission approval. Since the Commission has not reviewed the approaches to resolving these issues, they do not represent agency positions. Therefore, the staff regards these as open issues that must be satisfactorily resolved before it can complete its review of the Evolutionary Requirements Document. These issues will be closed once the Commission approves their resolution or provides alternative guidance.

Copies of SECY-90-016 and SECY-91-078, related ACRS reports and the staff's responses to those reports, and related Commission SRM are provided in Annexes A and B of this appendix. Annex C contains a copy of the February 27, 1992, draft Commission paper related ACRS reports, and the staff's response to one of those reports. Annex D contains a copy of the July 6, 1992, draft Commission paper.

Table 4B.1 Policy Issues for the Evolutionary Plant Designs

Policy Issues	Chapter	Section
Use of physically based source term	1	2.4.1.2
	1A	4.7
	1B	2.5.2
	1B	Annex A, Item I.B
	1B	Annex C, Item I.A
	5	2.7
Anticipated transients without scram	1B	3.2.16
	1B	Annex A, Item II.A
	1B	Annex C, Item I.B
	5	4.2, 4.3
Mid-loop operation	1B	Annex A, Item II.B
	1B	Annex C, Item I.C
	3	9
	5	5.2
Station blackout	1B	Annex A, Item II.C
	1B	Annex C, Item I.D
	5	2.2
	11	5
Fire protection	1B	Annex A, Item II.D
	1B	Annex C, Item I.D
	5	2.5
Intersystem loss-of-coolant-accident	1B	3.2.22
	1B	Annex A, Item II.E
	1B	Annex C, Item I.E
	3	9
	5	5.2
Hydrogen control	1	2.5.3
	1B	3.2.26
	1B	Annex A, Item III.A
	1B	Annex C, Item I.F
	5	2.3, 6.5, 6.6
Core-concrete interaction - capability to cool core debris	1B	Annex A, Item III.B
	1B	Annex C, Item I.G
	5	6.6.2
	6	4.3.2
High-pressure core melt ejection	1B	Annex A, Item III.C
	1B	Annex C, Item I.I
	5	7.2
Containment performance	1B	Annex A, Item III.D
	1B	Annex C, Item I.J
	5	2.1

Table 4B.1 (Continued)

Policy Issues	Chapter	Section
Dedicated containment vent penetration	1B	2.5.3
	1B	Annex A, Item III.E
	1B	Annex C, Item I.K
	5	6.6.3
Equipment survivability	1	4.8.2
	1B	Annex A, Item III.F
	1B	Annex C, Item I.L
	5	6.6.6
Elimination of operating-basis earthquake	1	4
	1B	2.1.1, 3.3.1
	1B	Annex A, Item IV.A
	1B	Annex C, Item I.M
	1B	Annex D, Item C
Inservice testing of pumps and valves	1	12
	1B	Annex A, Item IV.B
	1B	Annex C, Item I.N
	5	3.1
Industry codes and standards	1B	2.1.1
	1B	Annex C, Item II.A
Electrical distribution	1B	Annex B
	1B	Annex C, Item II.B
	11	4
Seismic hazard curves	1A	3.3
	1B	Annex C, Item II.C
Leak before break	1	4.5.5
	1B	Annex C, Item II.D
Classification of main steamline of boiling-water reactor	1B	2.3.1.1
	1B	3.2.8
	1B	Annex C, Item II.E
	2	3.4.1.5
	3	5.3.3, 5.4
	13	3.1.1
Tornado design basis	1	4.5.2.5
	1B	2.1.2
	1B	Annex C, Item II.F
Containment bypass	1A	4.4
	1B	Annex C, Item II.G
	5	4.5
Containment leak rate testing	1B	2.5.1
	1B	Annex C, Item II.H
	5	2.1.3, 6.3

Table 4B.1 (continued)

Policy Issues	Chapter	Section
Postaccident sampling system	1B 1B 3	2.3.2 Annex C, Item II.I 7
Level of detail	1B	Annex C, Item II.J
Prototyping	1B	Annex C, Item II.K
Inspections, tests, analyses, and acceptance criteria	1 1B	10 Annex C, Item II.L
Reliability assurance program	1 1B	6 Annex C, Item II.M
Probabilistic risk assessment beyond design certification	1A 1B 1B	All Annex C, Item II.N Annex D, Item E
Severe-accident mitigation design alternatives	1B	Annex C, Item II.O
Generic rulemaking related to FDA/DC	1B	Annex C, Item II.P
Defense against common-mode failures in digital instrumentation and control systems	1B	Annex D, Item A
Analysis of external events beyond the design basis	1A 1B	3 Annex D, Item B
Control room annunciator reliability	1B	Annex D, Item G

ANNEX A

DOCUMENTS RELATED TO SECY-90-016 "EVOLUTIONARY LIGHT WATER REACTOR CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS" JANUARY 12, 1990

CONTENTS

January 12, 1990

POLICY ISSUE
(Notation Vote)

SECY-90-016

For: The Commissioners

From: James M. Taylor
Executive Director for Operations

Subject: EVOLUTIONARY LIGHT WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Purpose: To present the staff's recommendations concerning proposed departures from current regulations for the evolutionary ALWRs. The staff requests Commission approval of the positions as described in this paper.

Background: In the April 21, July 31, and August 24, 1989 staff requirements memoranda (SRMs), the Commission asked the staff to identify the issues and acceptance criteria used to judge the acceptability of future designs; to identify where the staff proposes to go beyond the regulations or to be less restrictive; and to identify if the Advanced Boiling Water Reactor (ABWR) would meet the Commission's Safety Goal with or without a vent. The Commission asked that these issues be discussed in the context of certification of the ABWR and the other evolutionary advanced light water reactor (ALWR) designs as well as the staff's review of the evolutionary Electric Power Research Institute (EPRI) Requirements Document.

Discussion: Operating experience as well as a number of studies (e.g. PRAs) have identified a number of issues significant to reactor safety. Based on this background the staff has identified the following list of issues as fundamental to agency decisions on the acceptability of evolutionary ALWR designs.

(1) evolutionary LWR public safety goals
(2) source term

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(3) anticipated transients without scram
(4) mid-loop operation
(5) station blackout
(6) fire protection
(7) intersystem LOCA
(8) hydrogen generation and control
(9) core-concrete interaction - ability to cool core debris
(10) high pressure core melt ejection
(11) containment performance
(12) ABWR containment vent design
(13) equipment survivability
(14) operating basis earthquake/safe shutdown earthquake
(15) inservice testing of pumps and valves

The resolutions proposed by EPRI and the LWR vendors, and the staff positions and recommendations regarding each of these issues are discussed in detail in the enclosure. In addition to these issues, each application for a Design Certification will have to propose technical resolutions for those Unresolved Safety Issues and medium- and high-priorty Generic Safety Issues which are identified in NUREG-0933 and technically relevant to the design in accordance with 10 CFR 52.47 and the Severe Accident Policy Statement.

The Commission's approval of, or alternate guidance on, the proposed resolution of these issues is necessary for the staff's continued review of EPRI's ALWR Requirements Document, General Electric's (GE's) ABWR, Westinghouse's RESAR SP/90 and Combustion Engineering's (CE's) System 80+ designs. Approval or guidance is particularly important to the staff's evaluations of the GE ABWR and the EPRI ALWR Requirements Document since these reviews have progressed the furthest. The certification review for CE's System 80+ is just beginning. Westinghouse has indicated that they do not intend to pursue an FDA/certification for the RESAR SP/90 at this time. Additional Commission approval or guidance on significant issues related to certification of CE's System 80+ and other future designs will be discussed with the Commission as part of the development of the licensing review bases (LRB) for these designs. This approach is consistent with recent Commission guidance in an SRM dated December 15, 1989. It should be noted that some of the issues presented in the enclosure are issues proposed by EPRI which they refer to as plant optimization subjects. NRC approval of a plant optimization subject

would result in a resolution that is less restrictive than present regulations, Commission policy, or past licensing practices. For these reasons, the optimization subjects such as hydrogen control, source term, and the relationship between the operating bases earthquake and the safe-shutdown earthquake could have a major schedular impact on the evolutionary LWR certification process. The staff has provided the respective applicant's proposed solutions as well as the staff's positions on these issues in the enclosure to provide a comparison, and to provide an indication of the diversity of proposed solutions under consideration by the staff. The staff recommendations identified in this paper have been developed as a result of (1) the staff's reviews of current generation reactor designs and evolutionary ALWRs, (2) consideration of operating experience, including the TMI-2 accident, (3) results of the probabilistic risk assessments (PRAs) of current-generation reactor designs and the evolutionary LWRs, (4) early efforts conducted in support of severe accident rulemaking, and (5) research conducted to address previously identified safety issues. Information related to the issues and staff positions discussed in this paper have previously been provided to the Commission in SECY-89-013, SECY-89-153, SECY-89-228 and SECY-89-341 and have been underlined in the enclosure.

The staff positions recommended in this paper are consistent with those previously taken in the staff's review of the ABWR LRB and in several ABWR-related safety evaluation reports issued to date. The staff believes that, pending detailed staff review, there is a high degree of confidence that the ABWR would meet the positions identified in the enclosure. Therefore, Commission approval of the staff recommendations would close these policy issues for the ABWR and would permit staff review to continue. The recommended positions are also consistent with those identified in the staff's draft safety evaluations related to certain chapters of the EPRI-ALWR Requirements Document. The staff is reviewing severe accident and certification issues addressed in the EPRI-ALWR Requirements Document and the staff's final conclusions are awaiting Commission approval of the positions described in this paper. Approval of the staff recommendations would allow for continuation of the staff review of the EPRI ALWR Requirements Document in accordance with recent Commission guidance.

Conclusions: The staff believes its conclusions and recommendations regarding these matters are in keeping with the Commission's

policy expectation that future designs for nuclear plants will achieve a higher standard of severe accident safety performance.

The staff will inform the Commission during its reviews if additional enhancements to existing requirements, beyond those identified in the enclosure, are determined to be necessary for evolutionary ALWR designs.

Coordination: The Office of General Counsel has reviewed this paper and has no legal objection, but notes that any program for review of new reactor designs which authorizes the NRC to impose requirements beyond those needed to meet current Commission regulations raises the issue that, if the NRC staff can pose additional requirements for certification, other parties should be able to do so as well. The traditional way to avoid such problems is through rulemaking, but as indicated in SECY 89-311, the staff believes that design certification process is a more effective method of resolving severe accident issues than a generic severe accident rule or several individual changes. Further, OGC notes that questions regarding the desirability of additional severe accident mitigation measures still need to be addressed under NEPA, either in the design certification rulemaking and hearing or in some preliminary rulemaking.

A copy of this paper has been provided to the Advisory Committee on Reactor Safeguards.

Recommendations: That the Commission

(1) Approve the staff positions detailed in the enclosure, and

(2) Note that if the staff determines other issues need to be addressed in accordance with Commission guidance, the staff will inform the Commission of its positions on these matters in a timely manner.

(3) Note that the staff, in accordance with the Staff Requirements Memorandum dated August 24, 1989, plans to issue the draft SER on Chapter 5 of the EPRI ALWR Requirements Document.

(4) Note that following Commission resolution of the policy issues discussed in this paper, the staff plans to finalize and reissue the above draft SER.

James M. Taylor
Executive Director
for Operations

Enclosure:
As stated

Commissioners' comments or consent should be provided directly to the Office of the Secretary by COB Tuesday, January 30, 1990.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT Tuesday, January 23, 1990, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional time for analytical review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

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EVOLUTIONARY ALWR CERTIFICATION ISSUES

I. General Issues

A. ALWR Public Safety Goal

The EPRI Requirements Document proposes that the evolutionary ALWRs comply with the following public safety goals:

(1) The frequency of core damage will be less than 1.0 X 10E-5 events/per reactor-year, and (Note: EPRI refers to this as a "quantitative investment protection goal")

(2) Whole body dose at an assumed 0.5 miles site boundary must be less than 25 rem for events whose cumulative frequency exceeds 1 X 10E-6 per reactor-year.

In the Licensing Review Bases (LRB) for the ABWR, GE has committed to meet the following goals:

(1) Demonstrate that the likelihood of core damage will have a mean value of less than one in one hundred thousand reactor years (i.e., 1.0 X 10E-5).

(2) The expected mean frequency of occurrence of offsite doses in excess of 25 rem beyond a half mile radius from the reactor is to be less than once per million reactor years (i.e., 1.0 X 10E-6), considering both internal and external events.

(3) The containment design is to assure that the containment conditional failure probability is less than one in ten when weighted over credible core damage sequences.

The staff is presently reviewing the LRB for the System 80+ design in which CE has proposed goals which are similar to the goals developed in the ABWR LRB. Since Westinghouse has no immediate plans of pursuing certification of RESAR SP/90, work on the LRB is presently not planned. However, similar to GE and CE, Westinghouse has stated, in meetings with the staff and the Advisory Committee on Reactor Safeguards, they are committed to meeting the ALWR public safety goals as well as the goals in the Commission's Safety Goal Policy Statement.

The staff is reviewing the proposed ALWR public safety goals to ensure they are consistent with the Commission's Safety Goal Policy Statement, which proposed both qualitative as well as quantitative safety goals for future reactor designs. The current regulations do not specify requirements in numerical terms of frequency of core damage or large release events. However, the Commission in its Safety Goal Policy Statement, has proposed that the staff examine a general performance guideline that "the overall mean frequency of a large release of radioactivity to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation."

SECY-89-102 recommended approval by the Commission of the use of the following quantitative objectives in its implementation of the Safety Goal Policy for future standardized plants:

1. The mean core damage frequency target for each design should be less than 1.0 X 10E-5 event per reactor-year, and

2. The overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should 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 concludes that the staff-proposed quantitative safety goals submitted in SECY-89-102 are consistent with the Commission's Safety Goal Policy Statement. Additional Commission guidance on the establishment of quantitative goals and implementation of safety goal policy will assist the staff in its continuing assessment of the evolutionary ALWRs. Although the staff considers the goals defined in SECY-89-102 to be acceptable for evolutionary ALWRs, it should be noted that both the EPRI public safety goal and the ABWR public safety goal are considerably more stringent than the large release guideline defined in SECY-89-102.

Although the staff has indicated it believes the ALWR Public Safety Goal contains meritable goals for the industry to adopt, the staff has not completed its review of this issue and is in the process of reviewing how EPRI implements these goals.

B. Source Term

As noted in SECY 89-341, the staff's methodology for determining compliance with the siting requirements of 10 CFR Part 100 has been based on the 1962 "TID-14844" source term. This methodology, which involves calculation of offsite dose for comparison against Part 100 dose criteria (i.e. criteria for establishing the size of the exclusion area and the low population zone), is widely acknowledged to utilize conservative assumptions. At the time this approach was developed, these conservatisms were considered appropriate and were based on uncertainties associated with accident sequences and equipment performance; and as a means to assure that future plant sites would be essentially equivalent to sites approved up until that time. The conservatisms initially included in the methodology have been essentially retained up to this time.

EPRI has stated that the evolutionary ALWR licensing design-basis requirements as well as design enhancements related to severe accidents should be based on the full body of current knowledge regarding accident source terms. They believe that the evolutionary designs should be evaluated based on a realistic treatment of fission product source terms, including the extensive research that has been done on fission product behavior since TID-14844 was issued, and especially since the Three Mile Island accident in 1979. EPRI's view is that this approach will result in designs which are improved and provide enhanced safety protection. EPRI has identified this as a plant optimization issue.

GE has indicated that the ABWR will meet the offsite dose criteria established in 10 CFR Part 100. However, they propose to utilize updated information such as system performance and reliability information, developed since promulgation of Part 100, to justify some departure from the current methodology for calculating the offsite dose. The ABWR's current design includes a single stand-by gas treatment system (SGTS) charcoal filter bed, and no main steam isolation valve (MSIV) leakage control system (LCS). Previous BWR designs utilized redundant SGTS charcoal filter beds and, since 1976, most have been equipped with a MSIV-LCS. The staff's interpretation of the General Design Criteria (GDCs) would classify filters as active components and require redundancy to permit any dose reduction credit in calculating a Part 100 dose. Since 1976, MSIV-LCSs have been required in most BWRs to meet 10 CFR Part 100 for design bases accidents. Part 100 requires equipment used to mitigate consequences of design basis accidents to be seismically designed (it identifies equipment necessary to mitigate the consequences of accidents whose offsite consequences are comparable to the Part 100 dose guidelines as designed to withstand the vibratory motion of an SSE). Since non-safety grade equipment such as piping downstream of the MSIVs and the condenser are not seismically designed for SSE, credit for these systems has not been accorded in calculating offsite doses for Part 100 purposes. The staff is considering these deviations from the current methodology for demonstrating compliance with Part 100. The staff has concluded, based on current information and experience, that some deviation from current practice, or exemptions from the regulations identified above, may be warranted in the review of evolutionary designs. Presently, the staff believes that no other deviations would be necessary to demonstrate ABWR compliance with Part 100.

The other evolutionary ALWR vendors (Westinghouse and Combustion Engineering) have indicated that their evolutionary designs will comply with 10 CFR Part 100 and that they will work with the NRC and EPRI to utilize more realistic source term information to assess design enhancements related to severe accidents.

As stated in SECY 89-341, the staff is undertaking an examination of the implications of decoupling siting from plant design for future reactors. Under this plan, reactor site characteristics would be reviewed separately from the reactor without utilizing source terms or dose calculations. This would require revision to Part 100 and other regulatory staff practices. The results of such a study will establish appropriate guidelines for any future plant license applications. In the interim, however, the staff recommends that the Commission approve the following approach for evolutionary ALWRs:

- Assure that evolutionary designs meet the requirements of 10 CFR 100
- Consider deviations to current methodology utilized to calculate Part 100 doses on a case-by-case basis utilizing engineering judgement including updated information on source term and equipment reliability.

Such deviations could impact plant design features, therefore, these deviations will be identified in the SERs that will be forwarded to the Commission for its information well in advance of issuance, as directed in the SRM pertaining to SECY 89-311 dated December 15, 1989.

- Do not modify current siting practice, even though it is recognized that such deviations could result in calculated low population zones and exclusion areas which are smaller than those that have been approved for currently operating reactors.

- Continue to interact with EPRI and the evolutionary ALWR vendors to reach agreement on the appropriate use of updated source term information for severe accident performance considerations.

II. Preventative Feature Issues

A. Anticipated Transient Without Scram (ATWS)

The ATWS rule 10CFR 50.62 was promulgated to reduce the probability of an ATWS event and to enhance mitigation capability if such an event occurred.

EPRI has indicated that its approach to resolving the ATWS issue is compliance with the ATWS rule. Design requirements beyond those which would be required to meet the rule have not been proposed.

The ABWR design includes a number of features that reduce the risk from an ATWS event. These features include a diverse scram system with both hydraulic and electric run-in capabilities on the control rods, a manually operated standby liquid control system (SLCS), and a recirculation pump trip capability. In addition, the scram discharge volume has been removed from the ABWR, eliminating some of the potential ATWS problems associated with the older BWR designs. While the ATWS rule requires an automatically initiated SLCS, GE has concluded that the diverse scram system and enhanced reliability of the reactor protection system negates the need for an automatic SLCS. GE has agreed to provide a reliability analysis in order to support this position. The staff will review the analysis to determine if an exemption from 10 CFR 50.62, to approve manual SLCS, is justified. The staff analysis will be provided in a future safety evaluation report for the ABWR.

Westinghouse has concluded that a diverse scram system is unnecessary for the RESAR SP/90 design due to 1) high reliability of the integrated reactor protection system (IPS), 2) a turbine trip and emergency feedwater actuation that is independent of the IPS, 3) Ability to manually trip the rod control motor generators from the main control board, and 4) a highly negative moderator temperature coefficient. Westinghouse has committed to provide a detailed analysis to demonstrate that the consequences of an ATWS are acceptable at the time an FDA application is submitted.

The CE System 80+ design includes a control-grade Alternate Protection System which provides an alternate reactor trip signal and an alternate feedwater actuation signal separate and diverse from the safety-grade reactor trip system.

The staff believes, notwithstanding the Westinghouse position on diverse scram systems, that all future evolutionary ALWR designs should be required to provide a diverse scram system unless the LWR vendor can demonstrate that the consequences of an ATWS are acceptable. The ATWS rule presently requires a diverse scram system for all (CE, Babcock and Wilcox, and GE) LWR designs except Westinghouse PWRs. It had been determined that previous Westinghouse designs had adequate ATWS capability and backfit could not be justified. The staff believes that evolutionary ALWR designs should provide diverse methods of inserting control rods to mitigate a potential ATWS and to ensure a safe reactor shutdown. The staff considers that diverse scram capability is a worthwhile measure of prevention for all evolutionary ALWRs, especially when incorporated into the initial design. Imposition of a diverse scram system on the Westinghouse design would exceed the Commission's regulations. Therefore, the staff recommends that the Commission approve the staff position that diverse scram systems be provided for evolutionary ALWRs.

B. Mid-Loop Operation

The staff is concerned that decay heat removal capability could be lost when a PWR is shut down for refueling or maintenance and drained to a reduced reactor coolant system (RCS) or "mid-loop" level. For example, a significant problem has been the loss of residual heat removal (RHR) suction due to air-binding of the RHR pumps. This is usually caused by an uncontrolled low loop level and consequent air ingestion into the pump suction.

The EPRI Requirements Document specifies requirements consistent with measures applicable to operating reactors as described by the administrative procedures identified in Generic Letter 88-17, but does not specify design modifications to address the root cause of this event.

Westinghouse has committed to install a vortex breaker at the RHR hot leg connections to significantly reduce air entrainment during mid-loop operation. This feature, in conjunction with other design features of the plant, should greatly reduce concerns over mid-loop operation. CE has indicated that it will address this issue through analysis, consideration of specific design features, and/or operational restrictions. Specific design resolutions for the System 80+ have not been provided. Mid-loop operation is not an issue with the ABWR.

The staff expects improvements in instrumentation in many existing PWRs, but does not require specific modifications to the nuclear steam supply system (NSSS) to correct mid-loop problems. However, the staff believes that physical modifications such as those proposed by Westinghouse, may be necessary to essentially eliminate any concerns with mid-loop operation for future evolutionary pressurized ALWRs. Mid-loop operation is not explicitly covered by current regulations, however imposition of such requirements would exceed current staff licensing practices. Therefore, the staff recommends that the Commission approve the staff position that evolutionary PWR vendors propose design features to ensure high reliability of the shutdown decay heat removal system.

C. Station Blackout

The station blackout rule (10 CFR 50.63) allows utilities several design alternatives to ensure that an operating plant can safely shut down in the event that all ac power (offsite and onsite) is lost.

The EPRI Requirements Document provides for improvements in offsite power reliability, onsite power reliability and capacity, and station blackout coping capability. EPRI is also proposing that a large capacity, diverse alternate ac power source (combustion turbine generator) with the capability to power one complete set of normal safe shutdown loads be included in evolutionary ALWR designs.

The RESAR SP/90 emergency feedwater system includes two ac-independent and two dc-independent turbine-driven pumps. The electrical design includes two full capacity emergency diesel generators. In addition, it includes a backup seal injection pump powered by a small dedicated diesel generator which has enough capacity to also charge the station batteries. Westinghouse believes that this design will provide a 24-hour coping time which is sufficient to eliminate the need for the addition of an installed spare (full capacity) alternate ac power source.

The System 80+ design includes two turbine-driven emergency feedwater pumps and two motor-driven emergency feedwater pumps. The electrical design includes two full capacity emergency diesel generators and a diverse alternate ac power source. This alternate source of ac power is expected to be a control-grade combustion turbine with sufficient capacity and capability to power either one of the electrical divisions. In addition, the plant design has full load rejection capability and the capability to subsequently provide electrical power from the turbine generator. Each of the four safety-related instrument channels has a dedicated battery backup. Class 1E electrical Divisions I and II, which include the two emergency diesel generators are each provided dc power by an assigned pair of these batteries.

The ABWR design includes three independent electrical divisions, each with high-pressure and low-pressure water injection capability, each powered by a full capacity emergency diesel generator, and each division capable of independently shutting down the reactor. Additionally, the ABWR design includes an alternate ac combustion turbine to back up the diesel generators. The design has a capability to survive a 10-hour blackout period utilizing the reactor core isolation cooling (RCIC) turbine and station batteries. Extended blackout capabilities are also provided by the ac-independent water addition system. This system allows for makeup to the reactor vessel following RCS depressurization by connecting a direct drive diesel fire pump or by connecting an external pumping source, such as a fire truck, to a yard standpipe.

The staff believes that the preferred method of demonstrating compliance with 10 CFR 50.63 is through the installation of a spare (full capacity) alternate ac power source of diverse design that is consistent with the guidance in Regulatory Guide 1.155, and is capable of powering at least one complete set of

normal safe shutdown loads. Although an alternate ac power source is provided as an acceptable resolution to this issue in 10 CFR 50.63, staff imposition would exceed current Commission regulations. Therefore, the staff recommends that the Commission approve imposition of an alternate ac source for evolutionary ALWRs.

D. Fire Protection

The staff has concluded that fire protection 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 advanced plants, the staff concludes that current NRC guidance must be enhanced. Therefore, the evolutionary ALWR designers must ensure that safe shutdown can be achieved, assuming that all equipment in any one fire area will be rendered inoperable by fire and that re-entry into the fire area for repairs and operator actions is not possible. Because of its physical configuration, the control room is excluded from this approach, provided an independent alternative shutdown capability that is physically and electrically independent of the control room is included in the design. Evolutionary ALWRs must provide fire protection for redundant shutdown systems in the reactor containment building that will ensure, to the extent practicable, that one shutdown division will be free of fire damage. Additionally, the evolutionary ALWR designers must ensure that smoke, hot gases, or the fire suppressant will not migrate into other fire areas to the extent that they could adversely affect safe-shutdown capabilities, including operator actions. Because the layout of a nuclear plant is design-specific, plant-specific design details will be reviewed by the staff on an individual basis. The staff will require a description of safety-grade provisions for the fire-protection systems to ensure that the remaining shutdown capabilities are protected, as well as demonstration that the design complies with the migration criteria discussed above.

The ALWR Requirements Document indicates that fire protection will be as specified in 10 CFR 50.48 and Appendix R. It states that for equipment in the same general area, a 3-hour fire barrier will be utilized in lieu of physical separation unless it is "impractical or less safe." However, no guidelines are provided in the Requirements Document as to the application of these criteria.

The evolutionary ALWR designers have indicated that their fire protection designs are consistent with the staff's proposed enhancements. GE has provided its ABWR fire protection analysis which is currently under review by the staff.

Appendix R to 10 CFR Part 50 was promulgated for plants that were in operation prior to January 1, 1979. Subsequently, PRAs performed on more than a dozen plants have showed that fire is a significant contributor to core damage. The staff believes that in keeping with the Commission's desire for enhanced safety for evolutionary ALWRs, fire protection requirements should reflect experience from operating reactors and the greater understanding of severe accidents that has been acquired since Appendix R was promulgated. Therefore, the staff recommends the Commission approve the use of the enhanced fire protection position underlined above for evolutionary ALWRs.

E. Intersystem LOCA

Future evolutionary ALWR designs can reduce the possibility of a loss-of-coolant accident (LOCA) outside containment by designing (to the extent practicable) all systems and subsystems connected to the reactor coolant system (RCS) to an ultimate rupture strength at least equal to the full RCS pressure.

For both BWRs and PWRs, EPRI states that low-pressure systems which could be overpressurized by the RCS should be designed with sufficient margin to withstand full RCS pressure without structural failure.

For BWRs, pressure isolation valve instrumentation and controls should be provided to (1) prevent opening shutdown cooling connections to the vessel in any loop when the pool suction valve, discharge valve, or spray valves are open in the same loop, (2) prevent opening the shutdown connections to and from the vessel whenever the RCS pressure is above the shutdown range, (3) automatically close shutdown connections when RCS pressure rises above the shutdown range, and (4) prevent operation of shutdown suction valves in the event of a signal that the water level in the reactor is low.

For PWRs, relief valves sized to protect against overpressure transients, should be provided on the RHR system. RHR suction valves should be provided with permissive interlocks to prevent opening if RCS pressure exceeds RHR design pressure.

Westinghouse has indicated that, should the isolation valves of the RESAR SP/SC fail, the design pressure of the piping outside of the containment will be sufficient to withstand primary side pressure or will be vented to the Emergency water storage tank (EWST).

CE has eliminated the low-pressure safety injection system and increased the design pressure of the shutdown cooling system piping in the System 80+ design. With this higher design pressure, the shutdown cooling system is expected to maintain its integrity even when exposed to full reactor coolant system pressure.

The ABWR has been designed to minimize the possibility of an interfacing system LOCA in the following ways. The low pressure systems directly interfacing with the RCS are designed with 500 psig piping which provides for a rupture pressure of approximately 1000 psig. In addition, the high/low-pressure motor-operated isolation valves have safety-grade, redundant pressure interlocks. Also, the motor-operated emergency core cooling system (ECCS) valves will only be tested when the reactor is at low pressure. All inboard check valves on the ECCS will be testable and have position indication. Additionally, design criteria used by GE require that all pipe designed to 1/3 or greater of reactor pressure requires two malfunctions to occur before the pipe would be subjected to reactor system pressure. The pipe designed to less than 1/3 reactor pressure requires at least three malfunctions before the pipe would be subjected to reactor system pressure.

The staff concludes that designing, to the extent practicable, low-pressure systems to withstand full RCS pressure is an acceptable means for resolving this issue. However, the staff believes that for those systems that have not been designed to withstand full RCS pressure, evolutionary ALWRs should

provide (1) the capability for leak testing of the pressure isolation valves, (2) valve position indication that is available in the control room 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 attached low-pressure systems and both isolation valves are not closed. Imposition of these requirements exceed Commission regulations and guidance; therefore, the staff recommends that the Commission approve these positions for evolutionary ALWRs.

The staff notes that for some low-pressure systems attached to the RCS, it may not be practical or necessary to provide a higher system ultimate pressure capability for the entire low-pressure connected system. The staff will evaluate these exceptions on a case-by-case basis during specific design certification reviews.

III. Mitigative Feature Issues

A. Hydrogen Generation and Control

The Commission's Severe Accident and Standardization Policy Statements provide that future designs should address the provisions of 10 CFR 50.34(f). The Commission's stated policy has been codified in 10 CFR Part 52 to require the technically relevant provisions of 10 CFR 50.34(f) be met. Specifically, in order that containment integrity be maintained, 10 CFR 50.34(f)(2)(ix) requires future designs to provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100 percent fuel-clad metal-water reaction. In addition, the regulation requires this system to be capable of precluding uniform concentrations of hydrogen from exceeding 10 percent (by volume), or an inerted atmosphere within the containment must be provided.

The ALWR Requirements Document specifies that containment and combustible gas control systems should be designed to accommodate 75 percent in-vessel zirconium-water reaction of the active fuel cladding, and 13 percent containment uniform hydrogen concentration. It states that 75 percent cladding oxidation is believed to be a conservative upper limit on the amount of hydrogen generated in a degraded-core situation including recovery. EPRI has identified this as an optimization issue.

The RESAR SP/90 design proposes to mitigate the effects of a 100 percent metal-water reaction and to preclude uniform hydrogen concentration from exceeding 10 percent (by volume) through the use of hydrogen igniter and hydrogen recombiner systems.

The System 80+ design proposes to be consistent with the recommendations of the ALWR Requirements Document resulting from staff review. The information will include justifications for the assumed extent of metal-water reaction and the allowable uniform hydrogen concentrations.

The ABWR design meets the requirements of 10 CFR 50.34(f)(2)(ix) by utilizing a nitrogen-inerted atmosphere within its containment. Also, a hydrogen recombiner for design-basis accidents will be provided in the ABWR design.

Aside from the issue of regulatory compliance and applicability, and due to the uncertainties in the phenomenological knowledge of hydrogen generation and combustion, the staff concludes that compliance with the criteria of 10 CFR 50.34(f) remains appropriate for combustible gas control design in ALWRs. Research (discussed in NUREG/CR-4551) indicates that in-vessel hydrogen generation associated with core-damage accidents may range from approximately 40-95 percent active cladding oxidation equivalent. The amount of cladding oxidation is dependent on a variety of parameters related to sequence progression: reactor coolant system 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 ALWRs should provide protection for hydrogen generation resulting from a wider spectrum of accidents, i.e., full core-melt accidents with RPV failure. In that context, it is also 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 (1-2 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 due to the absence of 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 criteria and a reasonable surrogate for the combination of both in-vessel and ex-vessel hydrogen generation.

Due to the uncertainties in the phenomenological knowledge of hydrogen generation and combustion, it is still the staff's position that, as a minimum, evolutionary ALWRs should be designed to (1) accommodate hydrogen equivalent to 100-percent metal-water reaction of the fuel cladding and (2) limit containment hydrogen concentration to no greater than 10 percent. Furthermore, because hydrogen control is necessary to preclude local concentrations of hydrogen below detonable limits, and given uncertainties in present analytical capabilities, the staff concludes evolutionary ALWRs should provide containment-wide hydrogen control (e.g., igniters, inerting) for severe accidents. Additional advantages of providing hydrogen control mitigation features (rather than reliance on random ignition of richer mixtures) includes the lessening of pressure and temperature loadings on the containment and essential equipment. The staff recommends that the Commission approve the staff's position that the requirements of 10 CFR 50.34(f)(2)(ix) remain unchanged for evolutionary ALWRs.

B. Core-Concrete Interaction - Ability To Cool Core Debris

In the unlikely event of a severe accident in which the core has melted through the reactor vessel, it is possible that containment integrity could be breached if the molten core is not sufficiently cooled. In addition, interactions

between the core debris and concrete can generate large quantities of additional hydrogen and other non-condensible gases, which could contribute to eventual overpressure failure of the containment.

The EPRI Requirements Document contains a number of design features that are intended to mitigate the effects of a molten core. To promote long-term debris coolability, the Requirements Document states that the cavity floor should be sized to provide 0.02 m^2/MWt. The Requirements Document also specifies that the containment should be designed to ensure adequate water supply to the floor and that an alternate means of introducing water into the containment, independent of normal and emergency ac power, should be provided. Passive schemes for providing flooding of the floor areas beneath the vessel are proposed and described in general terms for both BWRs and PWRs. The Requirements Document also states that the steel shell or liner of the containment should be protected from core debris by at least 3 feet of concrete.

Westinghouse indicated that they will comply with the EPRI core-debris dispersal criteria of 0.02 m^2/MWt and that the RESAR SP/90 design will include some method (not yet defined) that would ensure automatic flooding of the lower cavity, using the in-containment refueling water storage tank (RWST), in the event of a severe accident. Westinghouse is currently evaluating alternative designs to ensure compliance with that commitment.

CE has also indicated that the System 80+ design will comply with the EPRI core-debris dispersal criteria of 0.02 m^2/MWt. Also, the in-containment refueling water storage tank will provide a source of water for lower cavity flooding.

The ABWR design has a number of features that the staff generally agrees would mitigate the effects of a molten core. It is designed with a lower drywell flooder and a cavity space sufficient to be able to disperse core debris at an energy level of 0.02 m^2/MWt. The flooder consists of a number of temperature-sensitive fusible plugs that allow suppression pool water to enter the drywell cavity when high temperature resulting from core debris occurs in the lower drywell. The horizontal vents to the suppression pool will remain covered in the event of lower drywell flooding, ensuring that releases continue to be scrubbed through the suppression pool water. GE anticipates that any core-concrete interaction will be stopped when the suppression pool water quenches the molten core debris. By providing sufficient area to allow the core debris to spread to a shallow bed and by flooding the core debris, it is expected that the potential for extensive core-concrete interactions will be significantly reduced. In addition, even if limited core-concrete interactions continue, the overlying pool of water will mitigate the consequences of these interactions by scrubbing the fission products and cooling the gases released from the core-concrete interaction.

The staff believes that an acceptable resolution to this issue can be provided by the evolutionary ALWR vendors if their designs

- provide sufficient reactor cavity floor space to enhance debris spreading, and

- provide for quenching debris in the reactor cavity.

Use of these criteria exceed current regulatory practice.

It should be noted that the specific cavity sizing criteria (0.02 m^2/MWt) proposed in the Requirements Document is still under evaluation by the staff. The issue of debris coolability is an area in which there is active ongoing experimental research including relatively large scale testing jointly sponsored by EPRI and NRC. Additionally, without assurance of core debris coolability, the level of protection afforded by a 3-foot thickness of concrete and the issue of vessel pedestal attack (ablation of concrete supporting the reactor vessel by the molten core debris) require further evaluation. The staff will continue to evaluate the issue of core debris coolability and the specific cavity sizing criteria (0.02m^2/MWt) proposed by EPRI as more data and information becomes available. The staff intends to assess the debris flooding schemes proposed by EPRI on a design-specific basis.

The staff recommends the Commission approve exceeding past regulatory practice in resolving this issue. The staff recommends approval of the general criteria, stated above, that evolutionary ALWR designs; 1) provide sufficient reactor cavity floor space to enhance debris spreading, and 2) provide for quenching debris in the reactor cavity. Design specific approaches to resolve this issue will be evaluated by the staff on a case-by-case basis to ensure compliance with these criteria.

C. High Pressure Core Melt Ejection

One potential effect of a severe accident that could potentially result in containment failure is the phenomenon of direct containment heating (DCH). The staff is concerned that this phenomenon might occur from the ejection of molten core debris under high pressure from the reactor vessel resulting in wide dispersal of core debris and extremely rapid addition of energy to the containment atmosphere.

To limit direct containment heating, the ALWR Requirements Document states that the cavity/pedestal/drywell configuration should be designed to preclude entrainment of core debris by gases ejected from a failed reactor vessel. It also states that a safety-grade RCS safety depressurization and vent system (SDVS) will be provided. The staff review has concluded that reactor vessel depressurization capability and cavity design features to entrap ejected core debris constitute an acceptable approach to the issue of high-pressure core melt ejection.

Westinghouse has indicated that the configuration of the cavity of the RESAR SP/90 containment will prevent core debris from entering the upper containment.

In addition, their ac independent depressurization system will reduce the probability of a high-pressure molten-core ejection from the reactor vessel.

CE has indicated the System 80+ design includes an indirect cavity vent path, including a debris collection chamber, (which is configured to de-entrain solid core debris and minimize direct containment heating) and a large floor area to enhance core debris coolability. In addition, the design includes a safety grade depressurization system which minimizes the possibility of high-pressure molten-core ejection.

The ABWR design incorporates a safety grade depressurization system and a suppression pool that surrounds the lower drywell cavity and thereby reduce the risk of high pressure core ejection and would prevent core debris from reaching the containment boundary and breaching its integrity.

The staff concludes that, during a high-pressure core-melt scenario, a depressurization system should provide a rate of RCS depressurization to preclude molten-core ejection and to reduce RCS pressure sufficiently to preclude creep rupture of steam generator tubes. Primary systems of evolutionary ALWRs should have the capability to be depressurized after loss of decay heat removal. In addition, the staff concludes that the ALWR Requirements Document should include a requirement that reactor cavities be arranged in such a manner that high-pressure core debris ejection resulting from vessel failure will not impinge on the containment boundary. The staff concludes that ALWR designs should include a depressurization system and cavity design features to contain ejected core debris. Imposition of these requirements exceed current Commission regulations. The staff recommends that the Commission approve this position for evolutionary ALWRs.

D. Containment Performance

The containment function, i.e., maintenance of a strong leak tight barrier against radioactive release, is faced with distinct challenges as a result of a severe accident. These challenges may be roughly divided into two categories, energetic or rapid energy releases and slower, gradually evolving releases to the closed containment system. Examples of containment loadings that fall into the first category include high-pressure core melt ejection with direct containment heating, hydrogen combustion, and the initial release of stored energy from the reactor coolant system. Slow energy releases to the containment are typified by decay heat and noncondensible gas generation. Engineering practice in containment design calls for passive capability in dealing with energetic energy releases where practicable while long-term energy releases may be controlled by both passive means as well as through active intervention.

In view of the low probability of accidents that would challenge the integrity of the containment, the staff concludes that the probability of failure of the mitigation systems (those systems which can reduce the consequences of a core damage accident), from the onset of core damage to loss of containment integrity resulting in an uncontrollable leakage substantially greater than the design basis leakage, should not exceed approximately 0.1. However, the staff intends

to ensure that the containment can deal with all credible challenges and does not intend to apply this conditional containment failure probability (CCFP) guideline in a manner that could be interpreted to potentially detract from overall safety. The staff will accept a CCFP of 0.1 or a deterministic containment performance goal that offers comparable protection. For this reason, the staff concludes that the following general criterion for containment performance during a severe-accident challenge would be appropriate for the evolutionary ALWRs in place of a CCFP.

> The containment should maintain its role as a reliable leak tight barrier by ensuring that containment stresses do not exceed ASME service level C limits for a minimum period of 24 hours following the onset of core damage and that following this 24 hour period the containment should continue to provide a barrier against the uncontrolled release of fission products.

Maintaining containment integrity for a minimum period (e.g. 24 hrs) is based on providing sufficient time for the remaining airborne activity in the containment (principally noble gases and iodine) to decay to a level that would not exceed 10 CFR Part 100 dose guideline values when analyzed realistically, if controlled venting were to occur after that time. During this period, containment integrity should be provided, to the extent practicable, by the passive capability of the containment itself and any related passive design features (e.g., suppression pool). The staff further believes that following this period, the containment should continue to provide a barrier against the uncontrolled release of fission products. However, in keeping with the concept of allowing for intervention in coping with long-term or gradual energy release, the staff believes that after this minimum period, the containment design may utilize controlled, elevated venting to reduce the probability of a catastrophic failure of the containment. Alternatively, a design may utilize diverse containment heat removal systems or rely on the restoration of normal containment heat removal capability if sufficient time is available for major recovery actions (e.g., 48 hours).

EPRI has indicated that the ALWR public safety criteria do not contain explicit criteria for conditional probability of containment failure or other mitigation features since the ALWR Steering Committee believes that such criteria could potentially distort the balance in safety design and inhibit innovative improvements in core protection features. However, EPRI has not yet indicated their position on an alternate containment performance goal.

Westinghouse has not yet committed to a specific containment performance goal for RESAR SP/90 although it is expected that the mitigation features discussed by Westinghouse would lead to a CCFP of less than 0.1 for all credible accident scenarios.

CE expects that the System 80+ design will meet the CCFP goal of 0.1 when weighted over credible core damage sequences given the following assumptions.

-Credible core damage sequences are defined as all core damage event sequences with a frequency of greater than 1.0 X 10E-6 per reactor year. External events which would cause both core damage and concurrently fail the containment and which have a frequency of less that 1.0 X 10E-5 per reactor year will not be considered.

-Containment failure is defined as a post core damage release resulting in a dose greater than 25 rem beyond one-half mile from the reactor.

The ABWR design currently includes a hardened wetwell vent for containment over pressure protection and is committed to a CCFP that is less than 0.1 when weighted over credible core damage sequences. In meetings with the staff, EPRI has stated that they consider a containment vent to be philosophically and institutionally undesirable and potentially unworkable. For additional information see related discussions under ALWR Public Safety Goal and ABWR Containment Vent Design.

Defense in depth, a long standing fundamental principle of reactor safety, results in the concept that multiple barriers should be provided to ensure against any significant release of radioactivity. In its Severe Accident Policy Statement, the Commission indicated that it "... fully expects that vendors engaged in designing new (or custom) plants will achieve a higher standard of severe accident safety performance than their prior designs." The Commission reaffirmed this policy in an SRM dated December 15, 1989 relating to SECY-89-311. A defense-in-depth approach reflects an awareness of the need to make conservative safety judgements in the face of uncertainties; in effect, not putting all the eggs in one basket. In that regard, the reactor containment boundary should serve as a reliable barrier against fission product release for credible severe-accident phenomena/challenges. Special effort should be made to eliminate or further reduce the likelihood of a sequence that could bypass the containment. The continued reliance on the traditional principle of containment of fission products following an accident is seen as a logical and prudent approach to addressing reasonable questions which will persist regarding the ability to accurately predict certain aspects of severe accident behavior. In order to ensure balance between prevention and mitigation, some criteria on containment performance are appropriate. Accordingly, a general goal of limiting the conditional containment failure probability to less than 1 in 10 when weighted over credible core-damage sequences would constitute appropriate attention to the defense-in-depth philosophy. Alternatively, a deterministic containment performance goal that provides comparable protection would be appropriate.

Probabilistic risk assessment (PRA) is a very powerful tool that permits systematic integrated assessment of design strengths and weaknesses. However, because very low frequency scenarios (approximately 1.0 X 10E-6 per reactor-year) are being addressed, it is important to recognize the large uncertainties in the quantification of these scenarios. The overall uncertainties in severe accident behavior are driven largely by insufficient data for assessing common-cause failures, difficulty in quantification of the potential for human

errors, and questions about completeness of analyses and uncertainties in phenomenological behavior. For this reason, the staff considers it acceptable to utilize a deterministic containment performance criterion that would provide a level of containment performance comparable to that which could be demonstrated using a probabilistic containment failure goal of 0.1, given a severe accident.

It is recommended that the Commission approve the staff's position to use a CCFP of 0.1 or a deterministic containment performance goal that offers comparable protection in the evaluation of evolutionary ALWRs.

E. ABWR Containment Vent Design

In Amendment 8 of the ABWR SSAR submittal (July 28, 1989) GE submitted a sensitivity analysis of the ABWR PRA to determine the net risk benefit of a vent system. Basically, this system is a containment overpressure relief system and is designed to avoid gross containment failure resulting from postulated slow rising overpressure scenarios that could result from postulated multiple safety system failures. These sensitivity analyses indicate that, with or without a vent system, the ABWR design meets the quantitative health objectives of the Commission's safety goal with a wide margin.

The staff's detailed review of the ABWR risk analyses, including the sensitivity analyses on the vent system, is currently underway. Based on the review to date, the staff believes that the scope of methods and data used in the ABWR PRA are sufficient and do not expect that the ABWR risk to exceed the Commission's quantitative health objectives with or without a vent system.

The staff's safety goal implementation plan also recommended that a subsidiary target related to plant performance be used. This target states that, for future plants, a mean core damage frequency due to internal events and external events be less than 1.0 X 10E-5 per reactor year of operation. The staff's review of Amendment 8 of the ABWR SSAR indicates that the overall core damage frequency from internal events (transients, ATWS events, and postulated LOCAs) and external events (primarily from beyond design basis seismic events and postulated fires) is about 6 X 10E-6 per reactor year. GE has determined that the proposed vent system has negligible impact on the core damage frequency. The staff notes that GE has provided an additional means of decay heat removal (a third train of RHR and an ac-independent water makeup system which relies on the fire water system to supply water to the core and containment sprays in emergency situations) for the ABWR design to reduce the frequency of the sequences involving loss of containment heat removal function, thus reducing the benefit (on core damage frequency only) of the ABWR vent system for these types of accident sequences.

The desirability of venting a BWR containment to mitigate multiple-failure accidents far beyond the design basis has been accepted for some time. Since 1981 the BWR Emergency Procedure Guidelines (EPGs), developed by the BWR Owners Group and approved by the NRC for existing BWRs, have called for venting the containment wetwell air space. GE believes containment

overpressure protection represents a practical and beneficial feature to incorporate in the ABWR. The overpressure protection feature is essentially passive, relatively inexpensive in a new plant, provides insurance against the consequences and financial risks associated with end-of-spectrum accident scenarios, is consistent with the BWR EPGs, and appears to be consistent with the ALWR philosophy of robustness.

GE has established two severe accident goals in the risk analyses submitted to the staff. These goals were defined in the ABWR LRB. The first goal states that the frequency of a severe accident release resulting in a whole body dose of 25 rem beyond one-half mile from the reactor should not exceed 1 X 10E-6 per reactor-year. This design goal is basically the same as the EPRI ALWR design goal. The second goal defined in the ABWR LRB states that the conditional containment failure probability should be less than one in ten (CCFP 0.1) when weighted over credible core damage sequences. The staff and GE agree that the definition of containment failure is an uncontrollable leakage substantially greater than the design basis resulting from loss of containment integrity following the onset of severe core damage. The ABWR design with the vent system is expected to meet the above goals; however, staff review in this area is not yet complete.

GE has performed an analysis utilizing this definition of containment failure to determine if the ABWR meets the CCFP goal of 0.1. The analysis indicates that the CCFP for the ABWR design, without a vent system, is equal to approximately 0.5 and does not meet the 0.1 goal, however with a vent system, the CCFP equals approximately 0.06.

Based upon the preliminary review of the ABWR severe accident design, the staff has determined that, as far as overall risk impact is concerned, the GE ABWR public safety goal is significantly more stringent than the Commission's quantitative health objectives. Also, the staff concludes that the public safety goal proposed by GE for the ABWR design is more stringent than the "large release guideline" as defined in the staff's proposed safety goal implementation plan. Therefore, based on the apparent enhanced level of safety provided by the ABWR's severe accident design features, which include the over pressure protection system, the staff recommends the Commission approve its use in the ABWR design certification process.

F. Equipment Survivability

With regard to the Commission's request concerning "The measures to ensure that systems and equipment required only to mitigate severe accidents are available to perform their intended function (e.g., environmental qualifications)," the staff believes that features provided for severe-accident protection (prevention and mitigation) only (not required for design basis accidents) need not be subject to (a) the 10 CFR 50.49 environmental qualification requirements, (b) all aspects of 10 CFR Part 50, Appendix B quality assurance requirements, or (c) 10 CFR Part 50, Appendix A redundancy/diversity requirements. The reason for this judgment is that the staff does not believe that severe core damage accidents should be design basis accidents (DBA) in the traditional sense that DBAs have been treated in the past.

Notwithstanding that judgment, however, mitigation features must be designed so there is reasonable assurance that they will operate in the severe-accident environment for which they are intended and over the time span for which they are needed. In instances where safety related equipment, (which is provided for design bases accidents) is relied upon to cope with severe accidents situations; there should also be a high confidence that this equipment will survive severe accident conditions for the period that is needed to perform its intended function. However it is not necessary for redundant trains to be qualified to meet this goal.

During the review of a specific ALWR design the credible severe accident scenarios, the equipment needed to perform mitigative functions, and the conditions under which the mitigative systems must function, will be identified. Equipment survivability expectations under severe accident conditions should include consideration of the circumstances of applicable initiating events (e.g., station blackout, earthquakes) and the environment (e.g., pressure, temperature, radiation) in which the equipment is relied upon to function. The required system performance criteria will be based on the results of these design-specific reviews. In addition, the staff concludes that severe-accident mitigation equipment for evolutionary ALWRs should be capable of being powered from an alternate power supply as well as from the normal Class 1E onsite systems. Appendices A and B to Regulatory Guide 1.155, "Station Blackout," provide guidance on the type of quality assurance activities and specifications which the staff concludes are appropriate for equipment utilized to prevent and mitigate the consequences of severe accidents.

The staff requests that the Commission approve the staff position that features provided only for severe-accident protection need not be subject to the 10 CFR 50.49 environmental qualification requirements, 10 CFR Part 50, Appendix B quality assurance requirements, and 10 CFR Part 50, Appendix A redundancy/diversity requirements.

IV. Non-Severe Accident Issues

The following issues, which are not normally considered through PRA analysis or not considered as severe accident issues for the evolutionary ALWRs, are brought to the Commission's attention because either the staff's positions or the vendor requests differ from past practices.

A. Operating Bases Earthquake(OBE)/Safe Shutdown Earthquake(SSE)

Presently, 10 CFR Part 100 requires that the magnitude of the OBE be at least one-half that of the SSE. It has been an industry wide experience that such a requirement leads to a design that is governed by the OBE requirements and produces unnecessary and inconsistent margins for the SSE loading. This requirement was included in the regulation when the staff did not have substantial experience with the seismic resistance of plants that incorporated OBE design at half the SSE value. Since then a number of research programs have been conducted including a large industry effort on testing and observation of actual earthquake experience of industrial facilities; consequently, the NRC funded Piping Review Committee has concluded that the OBE at existing plants are too high, therefore, controlling the design of some safety systems, and recommended that the OBE be decoupled from the SSE. Certain interim measures,

such as allowing somewhat higher damping values for piping analysis, have been taken to partially implement the Piping Review Committee recommendations (NUREG 1061, 1984). But the complete implementation of the recommendations would involve a revision of 10 CFR Part 100, Appendix A. Because of higher priority work, the effort on revision of this regulation has been postponed. It should be noted that the Commission has, in certain site specific cases, previously approved OBEs of less than one-half the SSE.

EPRI has requested that NRC regulations be changed to reduce the magnitude of the OBE relative to the SSE as a basis for the design. All evolutionary ALWR vendors agree with the request. GE has stated that they agree with EPRI in principle, however, the ABWR design uses an OBE that is one-half the SSE; therefore, this is a non-issue for the ABWR. EPRI has identified this as an optimization issue.

The staff agrees that the OBE should not control the design of safety systems. However, a staff position on this issue to be applied generically to all future designs has not yet been fully developed. For the evolutionary reactors, the staff will consider requests to decouple the OBE from the SSE on a design-specific basis. Such a decoupling would require an exemption to the Commission's regulations, therefore the staff recommends that the Commission approve this design-specific relief approach.

B. Inservice Testing of Pumps and Valves

The ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components" has been used to establish past testing requirements for ASME Code Class 1, 2, and 3 safety-related pumps and valves. These requirements provide certain information on the operational readiness of the components, but in general, do not necessarily verify the capability of the components to perform their intended safety function. It is the staff's judgement that the Code does not assure the necessary level of component operability that is desired for the evolutionary LWR designs. The staff believes that the following aspects of pump and valve testing and inspection are necessary to provide an adequate level of assurance of operability. The following provisions should be applied to all safety related pumps and valves and not limited to ASME Code Class 1, 2, and 3 components.

-Piping design should incorporate provisions for full flow testing (maximum design flow) of pumps, and check valves.

-Designs should incorporate provisions to test motor operated valves under design basis differential pressure.

-Check valve testing should incorporate the use of advanced non-intrusive techniques to address degradation and performance characteristics.

-A program should be established to determine the frequency necessary for disassembly and inspection of pumps and valves to detect unacceptable degradation which cannot be detected through the use of advanced non-intrusive techniques.

The staff has informed EPRI and the evolutionary ALWR designers of its concerns. No position has yet been expressed by these groups. Imposition of these procedures would exceed past licensing guidance; therefore, the staff recommends that the Commission approve these provisions for evolutionary ALWRs.

REPORT BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
ON SECY-90-016
"EVOLUTIONARY LIGHT WATER REACTOR
CERTIFICATION ISSUES AND THEIR RELATIONSHIP
TO CURRENT REGULATORY REQUIREMENTS"
APRIL 26, 1990

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

April 26, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: EVOLUTIONARY LIGHT WATER REACTOR CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

During the 358th, 359th, and 360th meetings of the Advisory Committee on Reactor Safeguards, February 8-10, March 8-10, and April 5-7 and 18-19, 1990, we discussed with representatives of the NRC staff the staff's positions and recommendations concerning the evolutionary light water reactor (ELWR) certification issues contained in SECY-90-016 (Ref. 1). During some of these meetings, we had the benefit of discussions with representatives of the Electric Power Research Institute (EPRI) and the General Electric Company. We also had the benefit of the documents referenced.

We were told by the staff that the positions for which they are seeking Commission approval are described in the underlined portions of the enclosure to SECY-90-016, entitled "Evolutionary ALWR Certification Issues." Unless indicated otherwise, our comments relate to these staff positions. Our comments and recommendations on the staff positions are given below.

I. GENERAL ISSUES

1. Evolutionary LWR Public Safety Goals

The NRC staff has concluded that the quantitative goals submitted for Commission consideration in draft SECY-89-102 (Ref. 2) are acceptable for ELWRs. The staff notes that both public safety goals in the EPRI ALWR Requirements Document (Ref. 3) and the ABWR Licensing Review Basis Document (Ref. 4) are considerably more restrictive than the large-release guideline defined in draft SECY-89-102. The staff further notes that additional Commission guidance on quantitative safety goals will assist the staff in its continuing assessment of ELWRs.

We believe, as stated in our previous reports (e.g., ACRS report on Key Licensing Issues Associated With DOE Sponsored Reactor Designs, dated July 20, 1988), that the Commission's Safety Goal Policy is appropriate guidance for regulatory decisions relating to ELWRs, other advanced reactors, and the operating plants. We regard it as not inappropriate that applicants should work to tighter standards when it serves their purposes, but we do not believe it is appropriate that the NRC should require such standards. In its Safety Goal Policy the Commission, in effect, said it would regulate to a level of safety that is adequate, not to the highest level that is possible.

2. Source Term

This issue is dealt with by a proposal to assure that evolutionary designs meet the requirements of 10 CFR 100 (Reactor Site Criteria). The requirements of this regulation include a limit on doses experienced by an individual at the exclusion area boundary, and at the boundary of the low population zone during the course of an accident. In calculating these doses, the instructions in 10 CFR 100 prescribe that the fission products released to the containment must be those which would be expected from accidents which "result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products." For plants currently operating, regulatory guides have delineated specific, but somewhat arbitrary, quantities of fission products that are acceptable to the staff in calculating the leakage from containment and the resultant doses at the specified boundaries.

In contrast, for the ELWRs, the staff proposes to explore the specification of a source term on a case-by-case basis, rather than using the arbitrary source term prescribed in the past. Since the issue of siting of these plants is not yet resolved, and since revisions to 10 CFR 100 are being considered, there may be no alternative to proceeding as the staff proposes, however awkward it may seem.

However, we can make no informed judgment concerning the appropriateness of the procedure until we know more about the criteria to be used in the selection of a source term, and the results of its application.

II. PREVENTATIVE FEATURE ISSUES

3. Anticipated Transients Without Scram (ATWS)

The staff recommends that the Commission approve the staff's position that diverse scram systems be required for the ELWRs.

It appears to us that a design that can ride out an ATWS without serious damage is feasible for PWRs and is preferable to a scram

system with diverse logic, which has a reliability calculable, at best, with large uncertainty. We recommend that the staff permit demonstration that the consequences of an ATWS are acceptable as an alternative to a diverse scram logic. The uncertainty in such a demonstration is probably considerably less than that in demonstrating that the contribution of an ATWS to risk is made acceptable by installation of a diverse scram logic system.

4. Mid-Loop Operation

We have been told previously of evidence that events initiated during mid-loop operations may be major contributors to risk in PWRs. However, shutdown operations are generally not accounted for in PRA studies, such as those reported in NUREG-1150 (Ref. 5), so the risks are not well quantified. For the operating plants, this issue has been dealt with through resolution of Generic Issue 99 (Improved Reliability of RHR Capability in PWRs). For the ELWRs, the staff recommends that PWR applicants propose design features to ensure high reliability of the shutdown decay heat removal system.

We agree with the staff's proposal, but recommend that more specific requirements be considered for mid-loop operation:

- Design provisions to help ensure continuity of flow through the core and residual heat removal system with low-liquid levels at the junction of the DHR system suction lines and the RCS
- Provisions to ensure availability of reliable systems for decay heat removal
- Instrumentation for reliable measurements of liquid levels in the reactor vessel and at the junction of the DHR system suction lines and the RCS
- Provisions for maintaining containment closure or for rapid closure of containment openings

5. Station Blackout

The Station Blackout Rule (10 CFR 50.63) requires that each light-water nuclear power plant licensed to operate must be able to withstand for a specified duration, and then recover from, a station blackout as defined in 10 CFR 50.2. This rule permits the utilities to submit alternative methods for coping with station blackout. This rule also states that a method based on an alternate ac power source, as defined in 10 CFR 50.2, will constitute an acceptable capability.

For the ELWRs, the staff recommends that the Commission require the

installation of an alternate ac power source as the only basis taken to demonstrate compliance with 10 CFR 50.63. The staff recommends that the alternate ac source have capacity to supply power for one safety train, including one complete set of normal safe shutdown loads, and that it be of diverse design. The alternate ac power supply must be designed to serve any safety train when needed, thereby serving as an additional backup power supply for the Class IE power supplies. The staff has stated that the diversity requirement will not preclude use of diesel generators, even though diesel generators are used for the Class IE emergency power supplies.

Although taken by itself this may seem to be desirable, it has not been demonstrated that it is required to conform to the safety goal. Nevertheless, we endorse the staff's recommendation.

6. Fire Protection

The staff concluded that the fire protection issues raised through operating experience and the Individual Plant Examination for External Events (IPEEE) Program (Ref. 6) must be resolved for the ELWRs. To accomplish this, the staff is proposing that the current NRC guidance for fire protection be enhanced as described by the staff during the March 27, 1990, meeting of our Subcommittees on Extreme External Phenomena and Severe Accidents. The enhancements proposed by the staff when combined with the requirements of 10 CFR 50.48 (Fire Protection) without exception and the guidance provided by the Standard Review Plan Section 9.5.1 (Fire Protection Program) should constitute an acceptable basis for prescribing fire protection features for the ELWRs.

The proposed enhancements represent a significant improvement in physical separation requirements and in the need to consider the effects of smoke, heat, and fire suppressant migration into other areas. In particular, redundant train separation is likely to be the most significant feature leading to reduced fire risk. We recommend that the proposed enhancements include separation of environmental control systems.

The fire-risk issues that were examined in the Fire Risk Scoping Study (Ref. 7), however, are not fully addressed in SECY-90-016. They should be.

We agree with the staff's recommendation for resolution of this issue with the above caveats.

7. Intersystem LOCA

The staff's position is that designing low-pressure systems to withstand full RCS pressure (to the extent practicable) is an acceptable means for resolving this issue. For those systems that

have not been designed to withstand full RCS pressure, the staff indicates that other measures will be required. We recommend approval of the proposed staff resolution, provided consideration is given to all elements of the low pressure piping system (e.g., instrument lines, pump seals, heat exchanger tubes, and valve bonnets).

III. MITIGATIVE FEATURE ISSUES

8. Hydrogen Generation and Control

The staff recommends that the ELWR designs provide a system for hydrogen control that can safely accommodate hydrogen generated by the reaction of water with 100% of the fuel cladding surrounding the active fuel. (Note: This is not 100% of the fuel rod cladding, nor does it include other metal in the core which could produce hydrogen if it were heated to a red heat in the presence of steam.) There is substantial uncertainty in establishing the amount of hydrogen that might be formed in a severe accident. We support the staff's recommendation.

The staff also recommends that the system be capable of precluding uniform concentrations of hydrogen greater than 10%. The EPRI ALWR Requirements Document specifies 13%. We are not aware of any experimental or analytical work that demonstrates that the detonation of hydrogen at the 10%, 13%, or some other level could damage the integrity of the containment and essential components. It is our impression that the effect, if any, is something that experts dealing with gas explosions can calculate with reasonable confidence. We suggest that the staff seek further technical information on possible effects, including stratification, before establishing a limit for the average hydrogen concentration.

9. Core-Concrete Interaction - Ability To Cool Core Debris

The staff proposes that the ELWR designs provide sufficient reactor cavity floor space to enhance debris spreading, and provide for quenching of the debris in the reactor cavity. Quantification of what constitutes sufficient reactor cavity floor space is still an open question, as is the means by which one quenches the core debris. The resolution of this issue will require engineering judgment as many of the physical processes are not fully understood. We agree with the staff's recommendation.

10. High-Pressure Core Melt Ejection

To cope with the possible effects of direct containment heating (DCH), the staff concludes ". . . that ALWR design should include a depressurization system and cavity design features to contain ejected core debris."

This is an extremely improbable event, and we see no need to require two modes of coping with the possibility. Either depressurization or cavity design provisions alone should be adequate. Because of possible safety benefits for other events, reliable depressurization is probably the preferred approach.

11. Containment Performance

The staff recommends that a containment performance guideline, expressed as a conditional containment-failure probability (CCFP) of 0.1, be used in evaluation of the ELWR designs. As an alternative, the staff proposes a deterministic performance goal that it believes would offer comparable protection.

We have previously recommended (ACRS Comments on An Implementation Plan For The Safety Goal Policy, dated May 13, 1987) such a quantitative guideline for containment performance as a part of the implementation of the Safety Goal Policy. However, this should be regarded as guidance to the NRC staff in its development of requirements for applicants. Merely passing on this guidance to applicants is not enough because the definition of CCFP is too imprecise. The deterministic performance criterion for containment systems suggested by the staff is also difficult to interpret.

We have undertaken an effort (ACRS report on Containment Design Criteria, dated March 15, 1989) to propose containment design criteria for future plants. But, as we said at the beginning of our study, we did not expect that it would directly affect the certification of the ELWR designs. This was, to some extent, because we recognized that our study would take some time to complete, but principally because the ELWR designs are now essentially complete and have been for some time.

We understand that the staff, assisted by the Brookhaven National Laboratory, is developing a regulatory guide that would serve as a basis for review of ELWR containment performance. We believe that the staff proposal will be adequate for ELWR review if it is supported by an appropriate regulatory guide developed on a timely schedule, and if it can be reasonably demonstrated that a containment that meets this guidance has a CCFP of not more than 0.1.

12. ABWR Containment Vent Design

During our April 5-7, 1990 meeting, we heard presentations from the staff and the General Electric Company regarding the staff's proposal that the Commission approve the use of severe accident design features that include a containment overpressure protection system in the ABWR design. We recommend that use of a containment overprotection system be approved subject to the results of the regulatory review.

13. Equipment Survivability

The staff recommends that features provided in the ELWR designs that are intended only for severe accident protection (prevention and mitigation) need not be subject to 10 CFR 50.49 (Environmental Qualification Requirements), 10 CFR 50, Appendix A (Redundancy and Diversity Requirements), and 10 CFR 50, Appendix B (Quality Assurance Requirements). However, the staff will require that mitigation features must be designed so there is "reasonable assurance" that they will perform their intended function in the severe accident environment and over the time span for which they are needed. Further, the staff proposes that at least one train of features provided for design basis accident protection, but also relied upon for severe accident protection, must be able to survive severe accident conditions for the time period that is needed to perform its intended function with "high confidence." In addition, the staff proposes to require that severe accident mitigation equipment be capable of being powered from an alternate power supply, as well as from the normal Class IE on-site systems.

To accomplish "reasonable assurance" and "high confidence," the staff will require that severe accident protective features use high quality industrial grade components which will be selected for the service intended and qualified by analysis or tests.

We endorse the staff's position. We note, however, that in this instance the staff's position includes much more than the underlined portions of the enclosure to SECY-90-016.

IV. NON-SEVERE ACCIDENT ISSUES

14. Operating Basis Earthquake (OBE)/Safe Shutdown Earthquake (SSE)

The staff states that it has not yet developed a position on this issue that can be applied generically to all future designs and recommends that the Commission approve a design-specific approach. We have no objection to the staff considering exemptions to the requirement that the OBE be at least one-half the SSE, where this can be justified. We note that this has been done in the past for 14 plants at 9 sites, but in each case using site-specific data. Other bases for justification may have to be provided for un-sited standard plant designs.

In the longer term, we recommend that the staff and the industry attempt to develop a position that can be defined generically. One approach worthy of study would be to abandon the use of two earthquake levels for the design of structures, systems, and components. Instead, the design could be based only on the SSE, with appropriate load factors and limit states, and a smaller but more likely earthquake could be established as a threshold for plant shutdown and inspection.

15. Inservice Testing of Pumps and Valves

The staff proposes that certain aspects of the testing and inspection of pumps and valves be enhanced to ensure the necessary level of component operability for the ELWR designs. We endorse the staff's proposal with the following clarification and additions:

- Although not stated explicitly, we were told during the March 7, 1990 meeting of our Subcommittee on Mechanical Components that the staff intends to apply the requirements of Generic Letter 89-10 (Ref. 8) to the ELWR plants as well. We endorse this intention.

- We recommend that the staff's requirement for full-flow testing capability be extended to other safety-related valves (e.g., MOVs) not just check valves. The requirement for flow testing of MOVs is included in Generic Letter 89-10.

- We recommend that the staff resolve the issue of check valve testing and surveillance requirements for existing LWR plants and indicate how it is to be applied to the ELWRs prior to issuing the FDAs.

- We recommend that the staff be encouraged to entertain proposals from the FDA applicants regarding alternative ways of meeting the in-service testing and surveillance requirements.

Additional comments by ACRS Members Harold W. Lewis and James C. Carroll and ACRS Members William Kerr, David A. Ward, and James C. Carroll are presented below.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

Additional Comments by ACRS Members Harold W. Lewis and James C. Carroll

Apart from one paragraph submerged as part of Item 1, this letter endorses the scattershot approach the staff has taken to the important question of regulation of new reactors. It therefore deserves to be called Camel II, in deference to the Committee's similar letter of January 15, 1987. The differences are that this list has in fact had more careful consideration, and that its elements originated with specific staff proposals. Indeed, in many

cases the genealogy can be traced to industry initiatives, and the staff is simply proposing to make mandatory those things that the industry has previously proposed to do on its own. None of this pays the slightest attention to the Commission's Safety Goal Policy, nor is there any hint of an effort to seize this opportunity to move the regulatory process in the direction of coherence and consistency. This is a pudding without a theme.

Let us then try to provide some perspective, since the Committee has chosen not to do so.

The Committee has often commented on the central role of the Safety Goals in providing a focus and objective for the body of regulation. Since this list sets the tone for the licensing of the next generation of light-water reactors, it is particularly important that its relation to Commission policies, especially the Safety Goal Policy, be clear. At the risk of repetition, we, and we believe the Committee, have never urged that specific regulatory decisions (such as these) be judged individually in the context of the Safety Goals, but only that the body of regulation be judged in that light. Individual decisions must still be made deterministically, with expertise and good judgment, but as part of a coherent overall body of regulation. Still, fifteen items come close to being a "body", and it is informative to see the role of the Safety Goals in the formulation of the staff recommendations. The Safety Goal policy, and other commission policies, are supposed to provide the glue that binds the whole structure together.

In effect, the staff says that it has proposed to the Commission a set of new safety goals (SECY-89-102), the Commission has not acted on them, either way, and therefore the staff will use them as if the Commission had approved. While we sympathize with the staff predicament, we think that is entirely inappropriate. The staff proposals include such things as a core-damage probability of 1E-5 per reactor-year, a "large accident" probability of 1E-6 per reactor year (with a bizarre definition of large accident), and a so-called conditional containment-failure probability. Not one of these has been approved by the Commission, yet the staff has used them in formulating its proposed policies on these items. It has rationalized this usurpation of power by asking for Commission action on SECY-89-102, and by stating that its own safety goals are "consistent" with those of the Commission. Of course any set of goals more stringent than yours will be consistent with your own, and acceptance of this argument will mean that the staff can regulate beyond your policies, more or less at will. That is precisely the situation your original goals were intended to foreclose. The Committee has often recommended that your Safety Goals be used as a final statement of "how safe is safe enough", not as a rigid minimum level of safety, beyond which the sky is the limit. Of course the industry may well have good reason to go further, but that is another matter.

In addition, as your own OGC has pointed out in SECY-90-016, this has the potential to open a Pandora's box, in which each party to a licensing proceeding may be able to claim the rights the staff claims--to insist on improvements beyond the rules. You will have to face this problem at some time, and the sooner the better.

We do not wish to understate the difficulty involved in translating a safety-goal policy into a workable body of regulation. The Committee has written you of its own recommendations for an organized approach to that problem, but we believe it can and should do more. Nuclear safety is not helped by letting that problem fester--the fact that it is difficult is no excuse for inattention. It is too much to expect regulation to be coherent and rational in the absence of an objective for that regulation.

We do think it was useful for the Committee to respond to your specific request for technical help on the fifteen questions posed, but you should recognize that this was done in the absence of a measuring rod. Each item was therefore judged on its own, and the Committee has turned its back on the opportunity to respond in a structured and coherent way. Any one of these items might have come out differently if it had been measured against an underlying rationale. In our view, the Committee has forfeited a chance to be of real service to both you and the public.

Additional Comments by ACRS Members William Kerr, David A. Ward, and James C. Carroll

By the "rulemaking" approach to design certification the Commission has sidestepped the development of revisions to regulations that would reflect knowledge gained from experience and research over the last ten or more years. As a result, important new requirements are being imposed on applicants through a variety of staff actions and reactions. This is a loosely controlled process in which major policy decisions are made without an appropriate intensity of review. Contributing to the lack of discipline is what we believe to be a serious ambiguity in the Commission's policy on advanced reactors. The Commission has said it expects future reactors to be safer. But, whether this is a mandate or simply an expectation that a maturing industry will produce safer plants is not clear. The staff has interpreted it as a mandate and has translated this into an unauthorized extension of the safety goals. This is despite the statement in NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants," published June 1988 (p. 4-1) that "the Commission expects but does not require enhanced safety margins other than those that may be required by the Safety Goal Policy." The Commission should not indefinitely postpone the development of a modern set of regulations. Only in this way will a proper balance be struck between adequate protection of the

public health and safety and the advantages to the public that can come from efficient development of the nuclear power option.

References:

1. SECY-90-016, memorandum dated January 12, 1990, from J. Taylor, Executive Director for Operations, NRC, to the Commissioners, Subject: Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements

2. Draft SECY-89-102, memorandum dated March 30, 1989, from V. Stello, Jr., Executive Director for Operations, NRC, to the Commissioners, Subject: Implementation of Safety Goal Policy

3. Electric Power Research Institute (EPRI), Advanced Light Water Reactor Requirements Document (Chapters 1 through 13), issued December 1987

4. General Electric Company, Advanced Boiling Water Reactor Licensing Review Basis Document, issued August 1987

5. U. S. Nuclear Regulatory Commission, NUREG-1150, "Severe Accident Risks: An Assessment for Five U. S. Nuclear Power Plants," Volumes 1 and 2, dated June 1989

6. Memorandum dated March 8, 1990 from W. Minners, Office of Nuclear Regulatory Research, NRC, to R. Fraley, ACRS, Subject: Proposed Generic Letter on Individual Plant Examination for Severe Accident Vulnerabilities Due to External Events (IPEEE) and Supporting Documents (Predecisional)

7. Sandia National Laboratories, "Fire Risk Scoping Study: Investigation of Nuclear Power Plant Fire Risk, Including Previously Unaddressed Issues," NUREG/CR-5088, published January 1989

8. Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," issued on June 28, 1989 to licensees for all power reactors, BWRs, PWRs, and vendors in addition to General Codes applicable to generic letters.

STAFF RESPONSE TO ACRS CONCLUSIONS REGARDING
EVOLUTIONARY LIGHT WATER REACTOR
CERTIFICATION ISSUES
(REGARDING SECY-90-016)
APRIL 27, 1990

APR 27 1990

MEMORANDUM FOR: Chairman Carr
Commissioner Roberts
Commissioner Rogers
Commissioner Curtiss
Commissioner Remick

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: STAFF RESPONSE TO ACRS CONCLUSIONS REGARDING EVOLUTIONARY LIGHT WATER REACTOR CERTIFICATION ISSUES

The ACRS sent a letter to the Commission addressing the subject issues. In general, the ACRS supported the staff positions, with additional comments and recommendations. The staff has evaluated these ACRS comments and recommendations and its evaluation is provided in the enclosure.

Original Signed By:
James M. Taylor

James M. Taylor
Executive Director
for Operations

Enclosure:
As stated

Enclosure A

EVOLUTIONARY LIGHT WATER REACTOR CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

1. Evolutionary LWR Public Safety Goals

The Advisory Committee on Reactor Safeguards (ACRS) stated, "We regard it as not inappropriate that applicants should work to tighter standards when it serves their purposes, but we do not believe it is appropriate that the NRC should require such standards."

The staff is accepting EPRI's proposed core damage frequency goal and their large release safety goal. We are not proposing to push them for more conservative safety goals, but the staff expects ELWR applicants to demonstrate that they have taken reasonable steps to achieve their stated goals.

2. Source Term

The ACRS stated that, "Since the issue of siting of these plants is not yet resolved, and since revisions to 10 CFR 100 are being considered, there may be no alternative to proceeding as the staff proposes, however awkward it may seem. However, we can make no informed judgment concerning the appropriateness of the procedure until we know more about the criteria to be used in the selection of a source term, and the results of its application."

The staff will keep the ACRS informed of our source term approach for ALWRs as we develop the details of its application during our reviews.

3. Anticipated Transients Without Scram (ATWS)

The ACRS stated that "It appears to us that a design that can ride out an ATWS without serious damage is feasible for PWRs and is preferable to a scram system with diverse logic, which has a reliability calculable, at best, with large uncertainty. We recommend that the staff permit demonstration that the consequences of an ATWS are acceptable as an alternative to a diverse scram logic."

The staff agrees with the ACRS comment that resolution of ATWS through a design that can ride out an ATWS is preferable to reliance on a diverse scram logic. This option was offered in the staff Draft Safety Evaluation Report for Chapter 5 of the EPRI Requirements Document and has been proposed by EPRI in Chapter 10. Whether the currently proposed diverse systems could be deleted would be based on the adequacy of this design analysis to demonstrate that an ATWS would not lead to serious damage.

4. Mid-Loop Operation

The ACRS stated that, "We agree with the staff's proposal, but recommend that more specific requirements be considered for mid-loop operation:

- Design provisions to help ensure continuity of flow through the core and residual heat removal system with low liquid levels at the junction of the DHR system suction lines and the RCS.

- Provisions to ensure availability of reliable systems for decay heat removal.

- Instrumentation for reliable measurements of liquid levels in the reactor vessel and at the junction of the DHR system suction lines and the RCS.

- Provisions for maintaining containment closure or for rapid closure of containment openings.

The staff agrees with the ACRS comments. In fact, past staff actions on this issue for operating reactors focused on most of these items. The staff plans to ensure that these factors are addressed in the ELWR designs.

5. Station Blackout

The ACRS stated that the staff recommended that an alternate diverse ac power source capable of powering one complete safety train of safe shutdown loads be installed in the ELWRs as the means used to meet the requirements of the Station Blackout Rule (10 CFR 50.63). The ACRS noted that the rule permits alternative methods for coping with station blackout, and cited 10 CFR 50.2 as defining the acceptable capability for the alternate AC source.

The ACRS also stated, "Although taken by itself, this may seem to be desirable, it has not been demonstrated that it is required to conform to the safety goal. Nevertheless, we endorse the staff's recommendation."

The alternate ac power source was identified as the preferred approach for meeting the Station Blackout Rule only in ELWRs because it offers several advantages over other approaches. It has the advantage of powering a complete complement of safe shutdown equipment, it can be utilized in other events in addition to station blackout (such as during mid-loop operation when one of the normal Class 1E standby power sources is out for maintenance or otherwise unavailable), it is essentially not time limited while providing power during a station blackout, and it provides for a uniform hardware based approach requiring little or no coping analyses and fewer specialized operating procedures.

6. Fire Protection

The ACRS supports the enhanced fire protection requirements recommended by the staff. Additionally, they recommend further enhancements including;

1) Required separation of Environmental Control System (HVAC systems) so that independent fire areas containing redundant safe shutdown trains are not serviced by the same ventilation train and,

2) That the 6 items identified in the fire risk scoping study be fully addressed for ELWRs.

The staff proposed requirement to consider the effects of smoke, heat, and fire suppressant migration may result in separate HVAC systems. However, other options may be available to the designer.

In reference to the ACRS recommendation that the 6 fire risk scoping study issues be fully addressed for ELWRS, it is the staff opinion that this is being done through the proposed enhancements and current regulatory efforts as indicated below.

1. Improved Analytical Fire Codes

The enhanced guidance eliminates redundant safety systems in the same fire area so it is not necessary to have a code to show that the redundant train in the same area is protected. This issue is moot for ELWRs with the enhanced criteria.

2. Control System Interactions

Current regulations require an independent safe shutdown capability in the event of loss of the control room. This is reiterated in the enhanced guidance for ELWRs.

3. Total Environment Equipment Survival

GDC-3 requires fire protection systems to be designed not to impair safe shutdown systems. Although in some instances fire system actuation has disabled a safety system, in no case have redundant systems been disabled.

4. Fire Barrier Qualification, and

5. Manual Fire Fighting Effectiveness

Both of these issues (4 and 5) are addressed in the current regulations. The ELWR designs must include information to demonstrate that appropriate maintenance exists for fire barriers and training exists for fire brigades.

6. Seismic/Fire Interactions

The fire risk scoping study concluded that this could be addressed by a walkdown. The staff will evaluate this issue as part of the PRA review and expects that a walkdown will be required as part of ELWR design certification.

7. Intersystem LOCA

The ACRS stated that, "We recommend approval of the staff resolution, provided consideration is given to all elements of the low pressure system, (e.g., instrument lines, pump seals, heat exchanger tubes, and valve bonnets.)"

The staff agrees with the ACRS comment and will consider all elements of the low pressure system to ensure that the system, to the extent practicable, will withstand full RCS pressure.

8. Hydrogen Generation and Control

The ACRS stated that the, "Staff recommends that the ELWR designs provide for hydrogen control that can safely accommodate hydrogen generated by the reaction of water with 100% of the fuel cladding surrounding the active fuel." The ACRS stated that, "We support the staff's recommendation."

The ACRS also stated, "We are not aware of any experimental or analytical work that demonstrates that the detonation of hydrogen at the 10%, 13%, or some other level could damage the integrity of containment and essential components. It is our impression that the effect, if any, is something that experts dealing with gas explosions can calculate with reasonable confidence. We suggest that the staff seek further technical information on possible effects, including stratification, before establishing a limit for the average concentration."

This staff recommendation is consistent with the reqirements set forth in 10 CFR 50.34(f). Although this section of the regulations was originally written for a select group of plants, 10 CFR Part 52 established these requirements to be a minimum standard for future plants. In this context, the requirement of 100% has become a surrogate to address both in-vessel and ex-vessel hydrogen production. It was not intended to be mechanistically based, but rather a design goal that appears to be reasonable in view of all the analyses performed to date. The staff has not uncovered any new evidence in this regard that would cause them to alter their view.

While it is true that there are no analyses that demonstrate that detonation loads are a threat to the containment or components within the containment, the staff believes that such an endeavor would be extremely complex and coupled with large degrees of uncertainty. The question of wave reflection and whether they could possibly reinforce within the containment is but one unknown affect. Therefore, the staff believes a prudent course of action is to assure that the likelihood of both global and local detonations is minimized.

To this end the staff has reviewed the available detonation test results and concludes that 10% represents a reasonable limit to assure non-detonable mixtures. It is felt that the true lower limit lies somewhere between 10% and 13%. Therefore, establishing a 10% criteria provides a degree of margin which is to address the issue of local detonations and the potential for stratification resulting in concentration gradients within the containment.

9. Core-Concrete Interaction--Ability to Cool Core Debris

The ACRS agreed with the staff's recommendation that each ELWR design provide sufficient reactor cavity floor space to enhance debris spreading and provide for quenching of the debris in the reactor cavity. The staff had indicated that considerable uncertainty remains on debris coolability and that they would continue to evaluate the issue as more experimental data and information became available. ACRS concurred in this evaluation and stated that "...resolution of this issue will require engineering judgment as many of the physical processes are not fully understood."

Experiments to assess coolability of water covered debris are underway at Argonne National Laboratory under joint EPRI/Industry/NRC support. The results of these experiments are expected to be available within one year.

10. High-Pressure Core Melt Ejection

The ACRS stated, "This is an extremely improbable event, and we see no need to require two modes of coping with the possibility. Either depressurization or cavity design provisions alone should be adequate. Because of possible safety benefits for other events, reliable depressurization is probably the preferred approach."

High pressure melt scenarios are dominant for some designs. The staff proposal is to provide a design concept with a degree of mitigation along with a certain amount of prevention. The RCS depressurization capability retains a degree of uncertainty. Such questions as the rate of depressurization and the cut-off pressure need to be addressed. As a result, we believe that a design can be developed with little or no added expense that can be effective in potentially eliminating the direct path to the containment. Since there are no major down sides, it appears to make sense to pursue this parallel approach of both depressurization and cavity design.

11. Containment Performance

The ACRS stated that, "We believe that the staff proposal will be adequate for ELWR review if it is supported by an appropriate regulatory guide developed on a timely schedule, and if it can be reasonably demonstrated that a containment that meets this guidance has a CCFP of not more than 0.1."

The staff believes that a containment performance guideline, either a conditional containment failure probability of 0.1 or a deterministic performance goal that offers comparable protection, is necessary to give the ALWR designers guidance on what is acceptable for containment performance under severe accident conditions. The alternative is to have a containment designed for loss-of-coolant accident (LOCA) and other design basis accident (DBA) conditions and then to evaluate its severe accident failure probability after it is designed.

The staff will keep the ACRS informed of our development of a regulatory guide in this area and of our progress in evaluating ALWR containments using the containment performance guideline as our review progresses.

12. ABWR Containment Vent Design

The ACRS stated that "We recommend that use of a containment overprotection system be approved subject to the results of the regulatory review."

The staff's review of the vent system design is underway. The purpose of this review is to insure that any downside of venting is minimized and that capability is provided to maintain control over the venting process.

13. Equipment Survivability

The ACRS states that, "We endorse the staff's position. We note, however, that in this instance the staff's position includes much more than the underlined portions of the enclosure to SECY-90-016."

The ACRS supports the staff position that equipment for severe accident mitigation does not need to meet the same quality standards as equipment for design basis accidents. Instead of 10 CFR 50.49, 10 CFR 50, Appendix A, and 10 CFR Appendix B, severe accident equipment need only be high quality industrial grade, selected and qualified by analysis or test for the intended service.

14. Operating Basis Earthquake (OBE)/Safe Shutdown Earthquake (SSE)

ACRS has no objection to the staff considering exemptions to the requirement that OBE be at least one half the SSE, where this can be justified. ACRS suggests: 1) other bases for justification may have to be developed

for unsited standard plant designs, 2) design be based only on the SSE with appropriate load factors and limit states, and 3) a smaller but more likely earthquake be established as a threshold for plant shutdown and inspection.

The staff accepts the ACRS suggestion which is very close to the design specific approach proposed by the staff.

15. Inservice Testing of Pumps and Valves

The ACRS endorses the staff's proposed approach while emphasizing the need to apply the requirements of Generic Letter 89-10 to ELWR plants. The ACRS also recommends that the staff resolve the check valve testing and surveillance issue and indicate how it is to be applied to ELWRs including consideration of industry proposed alternative ways of meeting inservice and surveillance requirements.

The staff agrees with the ACRS recommendations and will keep the ACRS informed of any new developments in this area.

STAFF REQUIREMENTS MEMORANDUM
"SECY-90-016 - EVOLUTIONARY LIGHT WATER REACTOR
CERTIFICATION ISSUES AND THEIR RELATIONSHIP
TO CURRENT REGULATORY REQUIREMENTS"
JUNE 26, 1990

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICE OF THE SECRETARY

June 26, 1990

MEMORANDUM FOR: James M. Taylor, Executive Director for Operations

FROM: for Samuel J. Chilk, Secretary

SUBJECT: SECY-90-16 - EVOLUTIONARY LIGHT WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIPS TO CURRENT REGULATORY REQUIREMENTS

This is to advise you that the Commission as detailed below has approved in part and disapproved in part the staff's recommendations in SECY-90-16.

I. General Issues.

A. ALWR Public Safety Goal.

The Commission (with Chairman Carr and Commissioners Roberts, Curtiss and Remick agreeing) has disapproved the use of 10^{-5} per year of reactor operation as a core damage frequency for advanced designs. As noted in the SRM on SECY-89-102 (dated June 15, 1990), the Commission supports the use of 10^{-4} per year of reactor operation as a core damage frequency goal. Although the Commission strongly supports the use of the information and experience gained from the current generation of reactors as a basis for improving the safety performance of new designs, the NRC should not adopt industry objectives as a basis for establishing new requirements. However, if the staff in applying the criteria of 10 CFR Part 52 (and in view of the uncertainties associated with PRA's) concludes that additional requirements are needed, based on our experiences with prior designs, in order to provide

NOTE: THIS SRM AND THE SUBJECT SECY PAPER WILL BE MADE PUBLICLY AVAILABLE 10 WORKING DAYS AFTER ISSUANCE OF THE SRM.

assurance that future designs will meet the Safety Goal Policy Statement, then the staff should provide those additional requirements to the Commission for consideration as they are identified.

Commissioner Rogers approved the staff's use of 10^{-5} as an expected design target for ELWR designers and endorsed a requirement that applicants be able to demonstrate that they have taken reasonable steps to reach these targets. However, he does not endorse those goals as an absolute requirement for approval of any specific design.

Consistent with the Commission's decision on SECY-89-102, the Commission approved the overall mean frequency of a large release of radioactive material to the environment from a reactor accident as less than one in one million per year of reactor operation. The Commission has not agreed on a definition of a large release and has requested a paper from the staff (See SRM from SECY-89-102). (RES) (Suspense: 9/28/90) 9000136

B. Source Term.

The Commission (with all Commissioners agreeing) has approved the staff's approach to the source term with the addition of the following element:

- On an expedited basis, incorporate appropriate changes to regulations, regulatory practices, and the review process resulting from source term research. (RES/NRR) - As appropriate - Pending Source Term Study Results. SECY paper due imminently. 8906208

II. Preventative Feature Issues.

A. ATWS.

The Commission (with all Commissioners agreeing) has approved the staff position. However, if the applicant can demonstrate that the consequences of an ATWS are acceptable the staff should accept the demonstration as an alternative to the diverse scram system. Commissioner Curtiss further believes that the staff should retain the flexibility to accept designs with non-diverse scram logic in those instances where it is demonstrated to the staff's satisfaction that the reliability of the scram function is such that the risk from ATWS is insignificant. (NRR)

B. Mid-Loop Operation.

The Commission (with all Commissioners agreeing) has approved the staff's proposed position, with the ACRS recommendation of April 26, 1990, that four additional specific requirements be considered for mid-loop operation. (NRR)

C. Station Blackout.

The Commission (with all Commissioners agreeing) has approved the staff's position that the evolutionary ALWR's have an alternate ac power source of diverse design capable of powering at least one complete set of normal shutdown loads. The staff should provide a clear definition of "diversity" so as to provide guidance on whether it means different types, different manufacturers, different models, etc. Commissioner Curtiss noted that, in his view, the clarification should focus on limiting common mode failure potential but need not go so far as to require completely different generator driver technologies (e.g. should not necessarily require both diesel and gas turbine driven generators). (NRR)

D. Fire Protection.

The Commission (with all Commissioners agreeing) has approved the staff's position on fire protection as presented in SECY-90-16 and supplemented by the staff's April 27, 1990, response to the ACRS comments. (NRR)

E. Intersystem LOCA.

The Commission (with all Commissioners agreeing) has approved the staff's position on intersystem LOCA provided that, as recommended by the ACRS, all elements of the low pressure system are considered (e.g. instrument lines, pump seals, heat exchanger tubes, and valve bonnets.) (NRR)

III. Mitigative Feature Issues.

A. Hydrogen Generation and Control.

The Commission (with all Commissioners agreeing) has approved the staff's position that the requirements of 10 CFR 50.34(f)(2)(ix) should remain unchanged for evolutionary plants. The staff should seek additional technical information, as suggested by the ACRS, and if reconsideration is warranted the Commission should be advised. (RES/NRR)

B. Core-Concrete Interaction--Ability to Cool Core Debris.

The Commission (with all Commissioners agreeing) has approved the staff's position. (NRR/RES)

C. High Pressure Core Melt Ejection.

The Commission (with all Commissioners agreeing) has approved the staff's position that the ELWR designs include a depressurization system and cavity design to contain core debris. The cavity design, as a

mitigating feature, should not unduly interfere with operations including refueling, maintenance, or surveillance activities. (NRR/RES)

D. Containment Performance.

The Commission (with all Commissioners agreeing) has approved, consistent with SECY-89-102, the use of a 0.1 CCFP as a basis for establishing regulatory guidance for the ELWRs. This objective should not be imposed as a requirement in and of itself. The use of the CCFP should not discourage accident prevention and the staff should review suitable alternative, deterministically-established, containment performance objectives providing comparable mitigation capability if submitted by applicants. Any such alternatives should be submitted to the Commission following staff review. (NRR/RES)

E. ABWR Containment Vent Design.

The Commission (with all Commissioners agreeing) has approved the staff's recommended use of the containment overpressure protection system on the ABWR, subject to the results of the comprehensive regulatory review which should fully weigh the potential "downside" risks with the mitigation benefits of the system. Staff should ensure that full capability to maintain control over the venting process is provided. (NRR)

F. Equipment Survivability.

The Commission (with all Commissioners agreeing) has approved the staff's position. (NRR)

IV. Non-Severe Accident Issue.

A. Operating Basis Earthquake (OBE)/Safe Shutdown Earthquake (SSE).
The Commission (with all Commissioners agreeing) has approved the staff's position. (NRR)

B. Inservice Testing of Pumps and Valves.

The Commission (with all Commissioners agreeing) has approved the staff's position as supplemented in their April 27, 1990, response to the ACRS comments. The Commission notes that due consideration should be given to the practicality of designing testing capability, particularly for large pumps and valves. (NRR/RES)

The Commission also agreed that in those cases where the staff proposed requirements depart from current regulations, consideration should be given to incorporating these requirements into the regulations. (See SRM dated May 27, 1990, M90053A).

Finally, the staff is encouraged to strive to sustain the level of attention and resources that have been devoted recently to the review process for the EPRI requirements document. The recent comments of the EPRI representatives at the June 4, 1990 Commission briefing suggest that such a commitment, if sustained, can be most beneficial in assisting EPRI and the NRC staff in our respective efforts to reach a common understanding on the key technical issues. (NRR)

cc: Chairman Carr
Commissioner Roberts
Commissioner Rogers
Commissioner Curtiss
Commissioner Remick
OGC
ACRS
IG
ASLBP
ASLAP

ANNEX B

DOCUMENTS RELATED TO
SECY-91-078
"CHAPTER 11 OF THE ELECTRIC POWER RESEARCH INSTITUTE'S (EPRI's) REQUIREMENTS DOCUMENT AND ADDITIONAL EVOLUTIONARY LIGHT WATER REACTOR (LWR) CERTIFICATION ISSUES"
MARCH 25, 1991

CONTENTS

POLICY ISSUE
(Notation Vote)

March 25, 1991 SECY-91-078

For: The Commissioners

From: James M. Taylor
Executive Director for Operations

Subject: CHAPTER 11 OF THE ELECTRIC POWER RESEARCH INSTITUTE'S (EPRI'S) REQUIREMENTS DOCUMENT AND ADDITIONAL EVOLUTIONARY LIGHT WATER REACTOR (LWR) CERTIFICATION ISSUES

Purpose: To inform the Commission of the staff's intent to issue the draft safety evaluation report (DSER) for Chapter 11 of the EPRI Requirements Document. Additionally to request Commission approval of staff recommendations concerning additional proposed changes to regulatory practice for the evolutionary advanced light water reactors (ALWRs).

Background: In the staff requirements memorandum (SRM) of December 15, 1989 pertaining to SECY-89-334, "Recommended Priorities for Review of Standard Plant Design," the Commission provided the following guidance to the staff:

> The SERs on the EPRI Requirements Document for both the evolutionary and the passive plant designs should be submitted to the ACRS for review and to the Commission for information and for review and approval of policy issues for which the Commission has not previously decided.

Further, in the SRM of June 22, 1990 pertaining to SECY-90-146, "Process, Schedule, and Resources For the Review of Evolutionary and Passive Advanced Light Water Reactors," the Commission directed the staff to follow the process presented in SECY-90-065, "Evolutionary and Passive Advanced Light Water Reactor Resources and Schedules." The staff included

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NOTE: TO BE MADE PUBLICLY AVAILABLE WHEN THE FINAL SRM IS MADE AVAILABLE.

the aforementioned guidance to SECY-89-334 as an element of the process described in SECY-90-065. In SECY-90-401, "Draft SER for Chapters 6, 7, 8, 9, 12, and 13 of EPRI's ALWR Documents for Evolutionary Reactor Plant Designs," the staff, following this guidance, informed the Commission that it would issue DSERs for Chapters 6, 7, 8, 9, 12, and 13 of the EPRI Requirements Document for evolutionary reactor plant designs. The staff issued these documents on January 15, 1991. This paper provides the DSER for Chapter 11, "Electric Power Systems." The remaining DSERs for the evolutionary design criteria, Appendix A to Chapter 1 and Chapter 10, are currently in preparation by the staff.

Discussion: Operating experience and a number of studies (e.g., probabilistic risk assessments (PRAs)) have identified a number of issues significant to reactor safety. In addition, in SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," the staff identified several policy issues that apply to future evolutionary ALWR designs and for which the Commission provided guidance in its SRM of June 26, 1990. The staff, in its continued review of the EPRI ALWR Requirements Document, has identified the following two additional issues:

1. alternate source of power for non-safety loads
2. connection of safety bus offsite power sources through non-safety buses

Enclosure 1 contains a detailed discussion of each of these issues and addresses those instances in which the staff positions differ from current regulatory requirements or in which the staff is substantially supplementing or revising interpretive guidance applied to currently licensed light water reactors (LWRs). In this enclosure, the staff also discusses the nature of the current regulatory requirement or interpretation, the positions of the ALWR vendors and of EPRI, the departure that the staff is proposing, and the basis for the proposed departure. To aid in identifying the staff's positions, the staff has underlined its positions and has cross-referenced them with the sections in the Chapter 11 DSER where they are discussed.

The staff developed the recommendations identified in this paper after (1) reviewing current generation reactor designs and evolutionary ALWRs, (2) considering operating experience, and (3) evaluating the results of the PRAs of LWRs. In addition, these positions are consistent with current design practices at recently licensed operating reactors.

To follow the process outlined in SECY-90-065, the staff will need to identify policy issues and bring them to the Commission for guidance before completing and distributing the DSERs.

However, the staff identified the issues discussed in this paper as it developed the DSER on Chapter 11 of the EPRI ALWR Requirements Document. Accordingly, the staff has enclosed the completed DSER (Enclosure 2) to provide the Commission additional information regarding these matters and to put the identified issues into their proper technical context. Additionally the staff believes that it would be beneficial to provide EPRI with the DSER in parallel with Commission review. The DSER would indicate that the previously identified policy issues are before the Commission for consideration. Distribution of the documents would expedite the review schedule by providing EPRI with a listing of staff identified open issues. Resolution of these issues will be addressed in the final SER on the EPRI Requirements Document.

Conclusions: The staff believes its conclusions and recommendations regarding these matters are in keeping with the Commission's policy expectation that future designs for nuclear plants will achieve a higher standard of safety performance.

The staff requests the Commission's approval of, or alternate guidance on, the proposed resolution of these issues in order to continue to review Chapter 11 of EPRI's ALWR Requirements Document for evolutionary plants and to perform the design certification of General Electric's ABWR, and Combustion Engineering's System 80+ designs.

By permitting the issuance of the DSER for Chapter 11 to EPRI, the Commission could expedite the review schedule while it considers these policy issues. In the DSER, the staff states that these policy issues are before the Commission for consideration. The staff will provide a regulatory departure analysis, based on Enclosure 1 of this paper, as Appendix C to the DSER on Chapter 11.

The staff will promptly inform the Commission during its reviews if it determines that additional enhancements to existing requirements, beyond those already identified, are necessary for evolutionary ALWR designs.

Coordination: The Office of General Counsel has reviewed this paper and has no legal objection. This paper is being forwarded to the ACRS for their review and comments.

Recommendations: That the Commission

(1) Approve the staff positions detailed in Enclosure 1

(2) Note that if the staff identifies other policy issues, the staff will inform the Commission of its positions in a timely manner

(3) Note that absent alternative Commission guidance, the staff will issue the enclosed DSER on Chapter 11 of the EPRI ALWR Requirements Document for evolutionary plant designs 10 working days after the date of this paper. The DSER will identify the two instances in which the staff is proposing to depart from current regulatory requirements and will state that the Commission is reviewing the basis for the approach that the staff is proposing and, accordingly, may determine that such issues involve policy questions that the Commission may wish to consider.

James M. Taylor
Executive Director
for Operations

Enclosures:
1. Policy Issues Analysis and Recommendations
2. Draft Safety Evaluation Report on Chapter 11

Commissioners' comments or consent should be provided directly to the Office of the Secretary by COB Tuesday, April 9, 1991.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT Tuesday, April 2, 1991, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

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ENCLOSURE 1

POLICY ISSUES ANALYSIS AND RECOMMENDATIONS

1. Alternate Source of Power for Non-Safety Loads

General Design Criterion (GDC) 17, "Electric Power Systems," requires that an onsite electric power system and the offsite electric power system be provided to permit functioning of structures, systems, and components important to safety. The offsite electric power system must have two physically independent circuits from the transmission network to the onsite electric distribution system.

Although the NRC has not established regulatory requirements on the number of power sources to the following non-safety related loads, the licensees for almost all nuclear power plants in the United States have provided two power sources to nonsafety-related loads such as reactor coolant pumps, reactor recirculation pumps, main feedwater pumps, condensate pumps, and circulating water pumps. During unit operation, a fast transfer of the nonsafety loads is usually provided to the startup transformer when imminent loss of the unit auxiliary transformer is sensed, such as following a main generator trip or a fault of the unit auxiliary transformer circuit. This process maintains power to the identified nonsafety loads and allows the plant to be shut down under these circumstances without a loss of normal feedwater systems or of forced circulation to the reactor coolant system.

The Electric Power Research Institute (EPRI) has specified requirements in the Requirements Document that only provide for a single source of power from the unit auxiliary transformer to nonsafety loads such as reactor coolant pumps, reactor recirculation pumps, main feedwater pumps, condensate pumps, and circulating water pumps. This one power source provides power to these loads during power operation, startup, and shutdown. A generator circuit breaker will isolate the main generator from the unit auxiliary transformer circuit during startup and shutdown when the main generator is unavailable. In existing plant designs, the main generator circuit breaker/unit auxiliary transformer configuration provide this isolation. However, an alternate source of power to this group of nonsafety loads is also provided at recently licensed operating plants, but not in the criteria of the EPRI Requirements Document.

The General Electric (GE), Combustion Engineering (CE), and Westinghouse standard plant designs all use the main circuit breaker/unit auxiliary transformer configuration as the primary power source to the subject nonsafety loads. However, the Westinghouse SP/90 design also provides an alternate source of power to the nonsafety loads, and the GE ABWR design provides an alternate source of power to a portion of the nonsafety loads (one of four main nonsafety buses). The CE System 80+ design uses the EPRI approach (it does not provide for an alternate power source).

An additional source of power would significantly reduce the number of plant trips that involve a loss of power to the nonsafety loads and require that the plant be shut down under natural circulation. Such an additional source of power would improve plant safety, because these events continue to be identified as more severe than the turbine-trip-only event in standard plant safety analysis reports.

The staff concludes that EPRI should enhance the ALWR design criteria in this area because they are less conservative than those that have been provided in existing, recently licensed plant designs. Therefore, the staff's position is that an evolutionary ALWR design should include an alternate power source to the nonsafety loads unless the design can demonstrate that the design margins in the evolutionary ALWR will result in transients for a loss of nonsafety power event that are no more severe than those associated with the turbine-trip-only event in current existing plant designs.

The staff addressed this issue in Section 4.2.1 of the Chapter 11 DSER (Enclosure 2 of this paper).

2. Connection of Safety Bus Offsite Power Sources Through Nonsafety Buses

GDC 17 requires that an onsite electric power system and offsite electric power system be provided to permit functioning of structures, systems, and components important to safety. The offsite electric power system must have two physically independent circuits from the transmission networks to the onsite electric distribution system. Although GDC 17 specifies that two offsite power circuits are required, and although it provides further additional design criteria on these circuits, it does not specify whether the circuits should directly connect the safety buses to the offsite power sources or whether this connection could be made through intervening nonsafety buses. The Institute of Electrical and Electronic Engineers (IEEE) Standard 308-1974, which is endorsed by Regulatory Guide 1.32, Rev. 2, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants," provides a figure that shows the Class 1E safety buses are directly connected to an offsite power transformer (startup transformer). However, the figure is for illustration only, and the Standard does not require the direct connection. IEEE Standard 765-1983 allows the Class 1E safety buses to be connected through nonsafety buses to the offsite transformers, but it also states that direct connection of the two circuits to each redundant Class 1E bus may further improve availability. However, the staff has not endorsed IEEE Standard 765, with a regulatory guide. Therefore, no regulatory requirements or guidance address the connection of safety bus offsite power sources through nonsafety buses.

EPRI's position is that many current designs of U.S. and foreign plants feed safety buses through nonsafety buses or from common transformer windings, and operating experience with these designs has not indicated any particular shortcomings. EPRI stated that there are real benefits in not connecting the safety buses directly to the offsite power supply, such as better protection of Class 1E systems against voltage surges affecting the offsite source. EPRI also states that the ALWR design makes provisions for a direct connection between safety buses and the offsite source in the event of problems with the nonsafety buses through which the safety buses are fed. EPRI indicated that some circuits are manually actuated and directly connect the safety buses to the reserve offsite transformer. The staff accepts such circuits. However, the EPRI Requirements Document does not require such a feature.

In the GE ABWR standard plant design, all the offsite power sources are directly connected to the Class 1E safety buses with no intervening nonsafety buses. In the Westinghouse SP/90 design, one circuit provides a direct connection between the Class 1E safety buses and an offsite power source while the remaining circuits connecting the safety buses to the offsite sources are all routed through intervening nonsafety buses. The CE System 80+ design in this area is identical to the EPRI approach in that it provides for a direct connection between safety buses and the offsite source in the event of problems with the nonsafety buses through which the safety buses are fed.

The staff concludes that feeding the safety buses from the offsite power sources through nonsafety buses or from a common transformer winding with nonsafety loads is not the most reliable configuration. Such an arrangement increases the difficulty in properly regulating voltage at the safety buses, subjects the safety loads to transients caused by the nonsafety loads, and adds additional failure points between the offsite power sources and safety loads. Therefore, it is the staff's position that at least one offsite circuit to each redundant safety division should be supplied directly from one of the offsite power sources with no intervening nonsafety buses, in such a manner that the offsite source can power the safety buses upon a failure of any nonsafety bus.

The staff discusses this issue in Section 4.2.2 of the Chapter 11 DSER (Enclosure 2 to this paper).

ENCLOSURE 2 HAS BEEN SUPERSEDED BY

THIS REPORT AND, THEREFORE, IS

NOT INCLUDED IN THIS ANNEX

REPORT BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
ON SECY-91-078
"PROPOSED POLICY ISSUES IDENTIFIED IN SECY-91-078,
'CHAPTER 11 OF THE ELECTRIC POWER RESEARCH INSTITUTE'S (EPRI's)
REQUIREMENTS DOCUMENT AND ADDITIONAL EVOLUTIONARY
LIGHT WATER REACTOR (LWR) CERTIFICATION ISSUES'"
APRIL 23, 1991

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

April 23, 1991

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: PROPOSED POLICY ISSUES IDENTIFIED IN SECY-91-078, "CHAPTER 11 OF THE ELECTRIC POWER RESEARCH INSTITUTE'S (EPRI'S) REQUIREMENTS DOCUMENT AND ADDITIONAL EVOLUTIONARY LIGHT WATER REACTOR (LWR) CERTIFICATION ISSUES"

During the 372nd meeting of the Advisory Committee on Reactor Safeguards, April 11-13, 1991, we discussed the two Policy Issues identified in SECY-91-078 related to the certification of the Evolutionary Light Water Reactors. Our Subcommittee on Improved Light Water Reactors also discussed these issues on April 9-10, 1991 in its continuing review of the EPRI Advanced Light Water Reactors (ALWR) Requirements Document. During these meetings, we had the benefit of discussions with representatives of the NRC staff and EPRI. We also had the benefit of the documents referenced.

The staff's position regarding the first Policy Issue is that "an evolutionary ALWR design should include an alternate power source to the non-safety loads unless the design can demonstrate that the design margins in the evolutionary ALWR will result in transients for a loss of non-safety power event that are no more severe than those associated with the turbine-trip-only event in current existing plant designs." The staff's major concern is that the ALWR designs are departures from past practice and may result in an increased frequency of shutdowns that require cooling by natural circulation. Presently licensed plants have electrical systems that provide an alternate power source to non-safety loads on shutdown. However, the staff did not substantiate its concerns with respect to the proposed EPRI design requirements.

EPRI claims that the ALWR is designed to safely accommodate shutdown with natural circulation and that the increased frequency of such events is small with this design. The EPRI requirements for the ALWR electrical system design fully meet General Design Criterion (GDC) 17, "Electric Power Systems," and the staff guidance contained in Regulatory Guide 1.32, Revision 2, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants." The ALWR electrical power system design is arranged to

supply electric power to the plant's safety loads from the main generator, the plant switchyard, an independent transmission line, a gas turbine generator, and the diesel generators. The design uses a generator circuit breaker between the main generator and the step-up transformer and has an improved full turbine load rejection capability. EPRI claims high reliability of electric power to the unit auxiliary transformers and has provided data to support its claim that the benefits derived from adding an alternate power source to the non-safety loads are small and not cost effective. We concur with the EPRI position.

The staff's position regarding the second Policy Issue is based on a misunderstanding of the text of the EPRI requirements. As a result, the staff proposes an additional requirement that "at least one offsite circuit to each redundant safety division should be supplied directly from one of the offsite power sources with no intervening non-safety buses, in such a manner that the offsite source can power the safety buses upon a failure of any non-safety bus." The staff's concern is that routing offsite power to the safety buses through non-safety buses may subject safety equipment to undesirable disturbances on the non-safety buses. Therefore, the staff's position would require the capability to supply safety buses directly from offsite power. The staff did not substantiate its concern. However, the EPRI requirements for ALWR electrical power system design already provide one alternate circuit to each of the redundant safety divisions directly from offsite power. This meets the staff's position. EPRI agreed to clarify the text to document this requirement. EPRI's position is that the direct circuit from offsite to each of the redundant safety divisions should be the backup power supply and the normal supply should be from the plant's auxiliary electric system. We concur with EPRI's position, but do not believe that this should become a regulatory requirement.

Sincerely,

David A. Ward
Chairman

1. SECY-91-078, Memorandum dated March 25, 1991 for the Commissioners from James M. Taylor, Executive Director for Operations, Subject: Chapter 11 of the Electric Power Research Institute's (EPRI's) Requirements Document and Additional Evolutionary Light Water Reactor (LWR) Certification Issues (Predecisional)
2. U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Draft Safety Evaluation Report on Chapter 11 of

the Advanced Light Water Reactor Requirements Document for Evolutionary Plant Designs, March 1991

3. Electric Power Research Institute, "Advanced Light Water Reactor Requirements Document, Chapter 11 - Electric Power Systems," Issued April 11, 1989

STAFF RESPONSE TO ACRS CONCLUSIONS
REGARDING SECY-91-078
MAY 15, 1991

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

May 15, 1991

Mr. David A. Ward
Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Ward:

This letter is in response to your letter to Chairman Carr of April 23, 1991, in which you addressed the two policy issues that were identified in SECY-91-078 and that were also identified in the staff's draft safety evaluation report (DSER) on Chapter 11 of the Electric Power Research Institute (EPRI) Requirements Document for evolutionary plant designs.

The basis for the staff positions on these issues is derived from the lessons learned from operating experience coupled with the philosophy of defense in depth. As part of its review the staff must ensure that proposed changes in the design do not lead to increased likelihood of abnormal occurrences as compared to today's designs. The staff's positions on these issues are taken to ensure that, as a minimum, the levels of safety in these areas are no less than that found in many existing operating plant designs.

Regarding the first policy issue, the loss-of-power-to-non-safety-buses event continues to be identified as more severe than the turbine-trip-only event for current reactors. Also, most recently licensed plants in the United States have provided two power sources to non-safety loads. It should be noted that the loss of normal power to the non-safety buses is more complicated and places greater demands on hardware and personnel than simply a loss of feedwater event as shown in EPRI's presentation to the ACRS. The plant response to the event is significantly impacted by the unavailability of multiple major components. There is a loss of forced circulation by the reactor coolant pumps (recirculation pumps in BWRs), a loss of heat sink by the circulating water pumps, a loss of heat removal capability by the non-essential service water pumps, a loss of condensate feed by the condensate pumps, and a loss of heat removal capability by the component cooling water pumps. The staff believes that if the alternate power supply feature is not provided, the likelihood of this type of challenge per reactor year is increased by up to an order of magnitude. The addition of an alternate source of power to the non-safety buses is an important feature that significantly reduces the challenges to the plant presented by this complicated event.

The EPRI presentation also included a cost estimate ($15M ± 5M) for implementation of the recommended design features. This estimate appears to be high, and the staff will pursue this matter with EPRI in order to attain a more precise correlation between the specific design features involved and the estimated costs. It is the staff's opinion that a reasonable, simple design change providing all the recommended functions can be implemented at a fraction of the cost of the overall plant electrical system, and that this design change would constitute an important improvement in safety.

Regarding the second policy issue, the staff has taken the position that at least one offsite circuit to each redundant safety division should be supplied directly from one of the offsite power sources with no intervening non-safety buses, in such a manner that the offsite source can power the safety buses upon a failure of any non-safety bus.

As was stated in the DSER for Chapter 11, the text of the EPRI Requirements Document does not require this feature and the staff's position is that the EPRI Requirements Document should be amended to require it. EPRI did not disagree with this position. The DSER further states that it has been the staff's experience that the benefits to safety of not connecting safety buses through non-safety buses or to common transformer windings usually outweigh whatever safety benefits may be achieved by doing so.

These positions constitute policy issues for the Commission's consideration in accordance with the staff requirements memorandum of August 24, 1989, pertaining to SECY-89-228, "Draft Safety Evaluation Report on Chapter 5 of the ALWR Requirements Document," in which the Commission instructed the staff to:

> Identify these instances where the staff is proposing to depart from current regulations or where the staff is substantially supplementing or revising interpretive guidance applied to currently licensed LWRs.

In its staff requirements memorandum of December 15, 1989, pertaining to SECY-89-334, "Recommended Priorities for Review of Standard Plant Designs," the Commission also provided the following guidance to the staff:

> The SERs on the EPRI Requirements Document for both the evolutionary and the passive plant designs should be submitted to the ACRS for review and to the Commission for information and for review and approval of policy issues for which the Commission has not previously decided.

The staff's positions are not embodied in any current regulatory guidance, and in accordance with the Commission's desire to review such issues, the staff identified these issues to the Commission for their review and approval.

The staff believes that its recommendations regarding both of these matters will ensure that the likelihood of complex plant challenges is not increased and that the levels of safety in these areas are not reduced from that found in many existing operating plant designs. Therefore, the staff continues to recommend the Commission's approval of the staff's positions on these issues.

Sincerely,

Original Signed By:
James M. Taylor

James M. Taylor
Executive Director
for Operations

cc: Chairman Carr
Commissioner Rogers
Commissioner Curtiss
Commissioner Remick
SECY

STAFF REQUIREMENTS MEMORANDUM
"SECY-91-078 - CHAPTER 11 OF THE ELECTRIC POWER RESEARCH INSTITUTE'S (EPRI's) REQUIREMENTS DOCUMENT AND ADDITIONAL EVOLUTIONARY LIGHT WATER REACTOR (LWR) CERTIFICATION ISSUES"
AUGUST 15, 1991

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICE OF THE SECRETARY

August 15, 1991

MEMORANDUM FOR: James M. Taylor, Executive Director for Operations

FROM: Samuel J. Chilk, Secretary

SUBJECT: SECY-91-078 - CHAPTER 11 OF THE ELECTRIC POWER RESEARCH INSTITUTE'S (EPRI'S) REQUIREMENTS DOCUMENT AND ADDITIONAL EVOLUTIONARY LIGHT WATER REACTOR (LWR) CERTIFICATION ISSUES

This is to advise you that the Commission (with all Commissioners agreeing) has approved the staff's positions in SECY-91-078. The staff should proceed with the issuance of the DSER for Chapter 11 of the EPRI requirements document with the requirements that; 1) evolutionary ALWR plants include an alternate power source to the nonsafety loads to provide additional margin for abnormal events based on operational experience, and 2) a direct connection will be provided between the offsite power source and the safety buses that will function upon a failure of any nonsafety bus.

The Commission also has agreed that the staff should raise policy issues to the ACRS and Commission early in the process, so that the Commission may make a decision on the approach to be taken on such issues. This does not require the staff to wait for the completion of ready-to-issue DSERs or SERs before coming to the Commission with policy issues. Policy issues should be presented for Commission consideration at the earliest possible opportunity, following sufficient discussions with the applicant to flesh out the parameters of such issues for Commission resolution. In this regard, while the staff should engage in early and open discussions with industry in order to identify and clarify the scope of, and industry positions on, such policy issues, the Commission should be apprised of such issues and provide guidance and approval before the staff takes a position on such issues with the applicant or the industry.

SECY NOTE: This SRM, the subject SECY paper and the vote sheets of the Chairman and Commissioners Rogers, Curtiss and Remick will be made publicly available 10 working days from the date of this SRM.

On a related matter, the direction to provide draft SERs to the Commission for review is intended to permit the Commission an opportunity to review, at its discretion, such SERs prior to issuance and, at the same time, to permit issuance of draft SERs while the Commission proceeds in parallel with the resolution of any newly identified policy issues. The earlier the Commission is apprised of policy issues, the less likely it will be that the staff will have to issue draft SERs at a time when policy issues remain unresolved.

Also, the staff should consider incorporating any modifications that arise from Commission decisions that depart from current regulations or substantially supplement or revise interpretive guidance into the regulations and regulatory guidance.

cc: The Chairman
Commissioner Rogers
Commissioner Curtiss
Commissioner Remick
OGC
GPA
ACRS

ANNEX C

DOCUMENTS RELATED TO THE DRAFT COMMISSION PAPER "ISSUES PERTAINING TO EVOLUTIONARY AND PASSIVE LIGHT WATER REACTORS AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS" FEBRUARY 27, 1992

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DRAFT COMMISSION PAPER
"ISSUES PERTAINING TO EVOLUTIONARY AND PASSIVE
LIGHT WATER REACTORS AND THEIR RELATIONSHIP
TO CURRENT REGULATORY REQUIREMENTS"
FEBRUARY 27, 1992

DRAFT

For: The Commissioners

From: James M. Taylor
Executive Director for Operations

Subject: ISSUES PERTAINING TO EVOLUTIONARY AND PASSIVE LIGHT WATER REACTORS AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Purpose: To identify issues pertaining to evolutionary and passive light water reactors (LWRs) and the staff's recommendations concerning resolution of those issues for which the staff has completed its review. The staff requests that the Commission approve the positions described in this paper, and will inform the Commission of its proposed resolutions for the remaining issues upon completing its review of these items.

Summary: The staff has identified a number of policy and significant technical issues pertaining to the evolutionary and passive LWRs. This paper provides a brief description and status of issues previously identified to the Commission and new issues. The staff has underlined the positions for which it is requesting Commission approval. Issues for which no resolution has been proposed yet will be addressed in future communications with the Commission as the staff's review progresses.

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Background: The Commission instructed the staff to identify policy issues to the Advisory Committee on Reactor Safeguards (ACRS) and the Commission early in the review process, so that the Commission can make a decision on the approach to be taken for resolution of such issues. The staff was further instructed to provide an analysis detailing where it proposes departure from current regulations or where the staff is supplementing or revising interpretive guidance applied to currently licensed light water reactors.

The staff has forwarded several policy papers[1] to the Commission that identify proposed resolutions for policy matters and major technical issues concerning both the evolutionary (1300 MWe) and passive (600 MWe) LWR designs. SRMs have been issued to the staff that provide the Commission's decisions regarding resolution of certain policy and major issues for evolutionary LWR designs.

Discussion: The staff has identified a number of issues significant to reactor safety as it considered operating experience, probabilistic risk assessment studies, and evaluations of the evolutionary and passive LWR designs. In Commission papers SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," and SECY-91-078, "Chapter 11 of the Electric Power Research Institute's (EPRI's) Requirements Document and Additional Evolutionary Light Water Reactor (LWR) Certification Issues," the staff proposed resolutions for some of these issues for evolutionary designs, which the Commission addressed in their SRMs of June 26, 1990, and August 15, 1991, respectively. The staff also forwarded policy papers to the Commission that identify and address the status of the staff's review of other issues. For completeness, this paper provides a brief description and status of these issues as well as new issues that pertain to the Advanced Light Water Reactor (ALWR) designs.

In reviewing the EPRI's ALWR Requirements Document and the vendors' ALWR designs, the staff has identified the following policy and major technical issues for both the evolutionary and passive LWR designs. Certain issues represent matters specific to the design of these facilities while

[1]Enclosure 3 to this document lists the papers that the staff has forwarded to the Commission regarding policy issues that have been identified to date for both evolutionary and passive advanced light water reactors (ALWRs). The staff will reference applicable documents throughout this paper, as appropriate.

others address the implementation of the design certification process of 10 CFR Part 52.

These issues are addressed in Sections I, II, and III of Enclosure 1, as listed below:

I. SECY-90-016 Issues

A. use of physically-based source term
B. anticipated transients without scram (ATWS)
C. mid-loop operation
D. station blackout
E. fire protection
F. intersystem loss-of-coolant-accident
G. hydrogen control
H. core concrete interaction - ability to cool core debris
I. high pressure core melt ejection
J. containment performance
K. dedicated containment vent penetration
L. equipment survivability
M. elimination of operating basis earthquake (OBE)
N. in-service testing of pumps and valves

II. Other Evolutionary and Passive Design Issues

A. industry codes and standards
B. electrical distribution
C. seismic hazard curves and design parameters
D. leak-before-break
E. classification of main steamlines of boiling water reactors (BWRs)
F. tornado design basis
G. containment bypass
H. containment leak rate testing
I. post-accident sampling system
J. level of detail
K. prototyping
L. inspections, tests, analyses, and acceptance criteria (ITAAC)
M. reliability assurance program (RAP)
N. site specific probabilistic risk assessments
O. severe accident mitigation design alternatives (SAMDAs)
P. generic rulemaking related to design certification

III. Passive Design Issues Only

A. regulatory treatment of non-safety systems
B. definition of passive failure
C. thermal-hydraulic stability of the simplified boiling water reactor (SBWR)
D. safe shutdown requirements

E. control room habitability
F. radionuclide attenuation
G. simplification of off-site emergency planning

Enclosure 1 provides a discussion of the nature of the current regulatory requirement or interpretation, the positions of the ALWR vendors and of EPRI, and, where available, the resolution that the staff is proposing, including the departure, if any, from current regulatory requirements and practice, and the basis for the staff's position. To aid in identifying the staff's positions, the staff has underlined the positions for which it is requesting Commission approval. Issues for which no resolution has been proposed will be addressed in future communications with the Commission as the staff's review progresses.

Enclosure 2 is a list of Commission papers, sorted by issue, in which the staff addressed issues for the evolutionary, passive, and/or advanced designs. The staff is providing this list for reference while reviewing Enclosure 1.

The staff developed the recommendations identified in this paper after (1) reviewing current generation reactor designs, evolutionary designs, and limited passive ALWR design information, (2) considering operating experience, (3) evaluating the results of the PRAs of LWRs and ALWRs, and (4) considering the Commission's guidance on issues resolved for the evolutionary ALWRs. The staff concludes that these issues are fundamental to the agency's decisions on the acceptability of the evolutionary and passive designs. As discussed in Section II.P of Enclosure 1 and SECY-91-262, "Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light Water Reactor Designs," the staff proposes to implement final positions on these matters as approved by the Commission through individual design certifications and generic rulemaking, as appropriate, for the evolutionary and passive designs.

The staff has reviewed only conceptual design information from the passive plant vendors in conjunction with the EPRI Requirements Document for passive designs, so its positions regarding passive designs may change as detailed design information becomes available, and testing on the passive systems is completed.

The staff concludes that it would be beneficial to provide EPRI and the ALWR vendors with the staff's conclusions presented herein as expeditiously as possible in order to help bring to closure those matters concerning both the evolutionary and passive reactor designs, and to facilitate

further discussion of the issues for the passive designs. This would allow the vendors to further their designs.

Therefore, the staff proposes to issue Enclosures 1-3 of this paper to the industry after 3 working days from the date of this paper to support meetings between the NRC staff, EPRI, the vendors, and the ACRS. The staff will indicate that the proposed resolutions identified to the industry are before the Commission for consideration, and, therefore, may not represent final positions.

Schedule

In SECY-91-161, "Schedules for the Advanced Reactor Reviews and Regulatory Guidance Revisions," the staff included a separate milestone for identifying policy issues for both the evolutionary and passive LWRs to the Commission and ACRS. The staff is submitting this paper to complete this milestone for both the evolutionary and passive LWR designs.

The staff does not expect to identify additional policy matters for the evolutionary designs. However, as the staff proceeds with its review of the EPRI ALWR Requirements Document for passive plants and the passive LWR design applications, it may identify other issues not identified herein. The staff allowed time for identifying and addressing such issues in the schedule estimates for the passive reactor design reviews provided in SECY-91-161. The staff will notify the Commission of new policy matters when they are identified.

Conclusions: The staff requests approval of the proposed resolution of those issues pertaining to evolutionary LWR designs in order to complete its review of EPRI's ALWR Requirements Document for evolutionary plant designs and to perform the final design approval/design certification review of General Electric's Advanced Boiling Water Reactor and Asea Brown Boveri (ABB) Combustion Engineering's System 80+ designs.

The staff also requests the Commission to approve the proposed resolution of those issues pertaining to the passive designs. This will enable the staff to proceed with its review and to provide guidance to the reactor vendors in these areas early in the development of their passive designs. Issues for which the staff has not reached its final conclusions will be addressed in future papers to the Commission.

Coordination: The Office of General Counsel has reviewed this paper and has no legal objection. OGC notes that Commission approval would be tentative, subject to further review in design certification rulemakings, and communications of the Commission's positions to the ALWR vendors and EPRI should make this point. This paper is being forwarded to the ACRS for its review and comments.

Recommendations: That the Commission:

(1) Approve the staff's positions underlined in Enclosure 1.

(2) Note that the staff will seek Commission approval of its positions for those issues still under staff review before taking a final position with the applicant or industry.

(3) Note that the staff will provide Enclosures 1-3 to the ALWR vendors and EPRI after 3 working days from the date of this paper. The staff will indicate that the proposed resolutions are before the Commission for consideration, and, therefore, may not represent final positions.

(4) Note that if the staff identifies other policy issues, the staff will seek Commission approval of its positions in a timely manner.

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Enclosures:
1. Policy Issues Analysis and Recommendations
2. Commission Papers Applicable to ALWR Issues
3. References

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Enclosure 1

POLICY ISSUES ANALYSIS AND RECOMMENDATIONS

The staff has identified a number of issues significant to reactor safety as it considered operating experience, probabilistic risk assessment studies, and evaluations of the evolutionary (1300 MWe) and passive (600 MWe) LWR designs. In Commission papers SECY-90-016[1] and SECY-91-078, the staff proposed resolutions for some of these issues for evolutionary designs, which the Commission addressed in their SRMs of June 26, 1990 and August 15, 1991, respectively. The staff also forwarded policy papers to the Commission that identify and address the status of the staff's review of other issues. For completeness, this enclosure provides a brief description and status of these issues as well as new issues that pertain to both the evolutionary and passive LWR designs.

In reviewing the EPRI ALWR Requirements Document and the vendors' ALWR designs, the staff has identified the following policy and major technical issues for both the evolutionary and passive LWR designs. Certain issues represent matters specific to the design of these facilities while others address the implementation of the design certification process of 10 CFR Part 52. The following is a discussion of each of these issues. The staff has identified those instances in which its positions differ from current regulatory requirements or in which the staff is substantially supplementing or revising interpretive guidance applied to currently licensed LWRs. A discussion of the nature of the current regulatory requirement or interpretation, the positions of the ALWR vendors and of EPRI, and, where available, the resolution that the staff is proposing, including the departure, if any, from current regulatory requirements and practice, and the basis for the staff's position is provided for each issue. To aid in identifying the staff's positions, the staff has underlined the positions for which it is requesting Commission approval. Issues for which no resolution has been proposed will be addressed in future communications with the Commission as the staff's review progresses. Plant optimization subjects, which are items proposed by EPRI that do not meet regulatory requirements, are also identified.

This enclosure is divided into three sections. Section I addresses the staff's current positions regarding the applicability and resolution of issues identified in SECY-90-016 (for evolutionary designs only) for the passive LWR plant designs. Section II addresses the status and, where available, proposed resolution of other issues that are applicable to both the evolutionary and passive ALWR designs. Section III addresses issues that the staff has identified while reviewing the conceptual design information from the passive plant vendors in conjunction with the EPRI Requirements Document for passive designs. Positions on the passive designs may change as detailed design information becomes available and testing on the passive systems is completed.

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[1]Enclosure 3 lists the papers that the staff has forwarded to the Commission regarding policy issues that have been identified to date for both evolutionary and passive advanced light water reactors (ALWRs). The staff will reference applicable documents throughout this paper, as appropriate.

Enclosure 2 is a list of Commission papers, sorted by issue, in which the staff addressed issues for the evolutionary, passive, and advanced designs. This enclosure is included to provide a reference list for each issue. These papers will be referenced throughout this enclosure as appropriate.

I. SECY-90-016 ISSUES

The following is a discussion of the staff's current positions regarding the applicability and resolution of issues identified in SECY-90-016 (for evolutionary designs only) for the passive LWR plant designs. Unless specified otherwise, the implementation guidance provided in SECY-90-016 is considered applicable to the discussions below.

A. Use of Physically-Based Source Term

As discussed in SECY-90-016, the staff's methodology for determining compliance with the siting requirements of 10 CFR Part 100 has been based on the source term provided in Technical Information Document (TID)-14844, issued in 1962. This methodology, which involves calculation of offsite dose for comparison against Part 100 dose criteria (i.e., criteria for establishing the size of the exclusion area and the low population zone), is widely acknowledged to utilize conservative assumptions. At the time this approach was developed, these conservatisms were considered appropriate and were based on uncertainties associated with accident sequences and equipment performance, and as a means to assure that future plant sites would be essentially equivalent to sites approved up until that time. The conservatisms initially included in the methodology have been essentially retained up to this time.

EPRI has proposed a physically-based source term to be used for the licensing design basis fission product release based on a bounding severe reactor accident to be used for both the evolutionary and passive reactor designs. Technical justification is provided in their October 18, 1990 and February 12, 1991 submittals entitled "Licensing Design Basis Source Term Update for the Evolutionary ALWR," and "Passive ALWR Source Term," respectively. EPRI has identified this as a plant optimization subject.

The EPRI-proposed source terms are based on a single, enveloping value for a bounding severe reactor accident sequence, using release data obtained from the Severe Fuel Damage Tests at the Power Burst Facility, the LOFT source term measurements, and data from the TMI-2 post accident examination. For the evolutionary designs, EPRI proposes changes in the assumptions concerning the fission product fuel release magnitude, the fission product release timing, the chemical form of iodine, the retention of aerosol in the reactor coolant, and the use of the suppression pool and containment sprays for removal of aerosol and soluble gases. For the passive designs, EPRI proposes that the source term be also based on consideration of passive mitigation functions and systems such as steam condensation-driven aerosol removal, main steam isolation valve leakage control, and secondary building fission product leakage control.

In SECY-90-016, the staff recommended that the Commission approve the following approach for evolutionary ALWRs:

1. Assure that evolutionary designs meet the requirements of 10 CFR 100,

2. Consider deviations to current methodology used to calculate 10 CFR Part 100 doses on a case-by-case basis using engineering judgement including updated information on source term and equipment reliability,

3. Do not modify current siting practice, and

4. Continue to interact with EPRI and the evolutionary ALWR vendors to reach agreement on the appropriate use of updated source term information for severe accident performance considerations.

In its June 26, 1990 SRM, the Commission approved the staff's approach to the source term for the evolutionary designs with the addition that the staff incorporate appropriate changes to regulations, regulatory practices, and the review process resulting from source term research.

As a result of this guidance, the NRC staff is developing a revised source term based on source term calculations performed by the source term code package for individual accident sequences selected in NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," December 1990. The staff concludes that the fission product release source terms proposed by EPRI and those being developed by the staff are very close for all radionuclide groups, except tellurium and low volatile elements. The reasons for and the impact of the differences between Brookhaven National Laboratories' (BNL's) and EPRI's estimates for tellurium and the low volatile elements are still under review.

General Electric (GE) and Combustion Engineering, Inc. (CE) have indicated that their evolutionary designs (the Advanced Boiling Water Reactor (ABWR) and the System 80+, respectively) can meet the offsite dose criteria established in 10 CFR Part 100 using the current TID-14844 source term. However, the vendors have indicated that they may wish to apply the forthcoming updated TID source term to their designs when it becomes available. The staff concludes that this approach is acceptable provided the vendor addresses the updated TID source term package in its entirety. If the evolutionary vendors choose to adopt this source term, the FDA/DC application must include complete analyses of the radiological consequences of design basis accidents using all of the guidelines provided in the updated TID source term. However, this may impact the review schedules of the evolutionary LWRs, because the updated TID source term is currently scheduled to be published for a 90 day public comment period in April 1992, and the staff's technical position on the fission product removal mechanisms in the containment are not finalized.

The passive ALWR vendors (Westinghouse and GE) have indicated that their designs will comply with the source term developed as a result of the staff's

evaluation of this issue with EPRI. The effort to revise the TID source term is expected to be completed in a time period sufficient to support the FDA/DC review of the passive plant designs.

The staff is in the final stage of completing its proposed update of the TID-14844 source term, including fission product removal mechanisms within the containment. The revised source term and its applicability to the ALWRs will be addressed in a separate Commission paper scheduled in the first quarter of 1992. Therefore, the staff does not request any Commission action on this issue at this time.

B. Anticipated Transients Without Scram (ATWS)

As discussed in SECY-90-016, the ATWS rule (10 CFR 50.62) was promulgated to reduce the probability of an ATWS event and to enhance mitigation capability if such an event occurred. The staff recommended that the Commission approve its position that diverse scram systems should be provided for evolutionary ALWRs. In addition, the staff indicated that GE would perform a reliability analysis to determine whether they could justify the manual operation of the standby liquid control system (SLCS) in the event of an ATWS in lieu of automatic operation as required by 10 CFR 50.62.

In its June 26, 1990 SRM, the Commission approved the staff's position, but directed that, if the applicant can demonstrate that the consequences of an ATWS are acceptable, the staff should accept the demonstration as an alternative to the diverse scram system.

EPRI has indicated that its approach to resolving the ATWS issue is compliance with the ATWS rule. Design requirements beyond those which would be required to meet the rule have not been proposed. In its December 6, 1991 letter, EPRI stated it has determined that automatic actuation of the SLCS was appropriate, and that it was modifying the Requirements Document for evolutionary designs to reflect that position. EPRI no longer considers this to be a plant optimization subject.

The ABWR design includes a number of features that reduce the risk from an ATWS event. These features include a diverse scram system with both hydraulic and electric run-in capabilities on the control rods, a standby liquid control system (SLCS), and a recirculation pump trip capability. In addition, the scram discharge volume has been removed from the ABWR, eliminating some of the potential ATWS problems associated with the older BWR designs. In its letter dated October 9, 1991, GE indicated it will include an automatic SLCS.

The CE System 80+ design includes a control-grade alternate protection system which provides an alternate reactor trip signal and an alternate feedwater actuation signal separate and diverse from the safety-grade reactor trip system.

The staff is in the process of evaluating the evolutionary plant submittals to ensure acceptable implementation of the Commission's directive, but does not expect any related policy matters to result from its review.

Although EPRI has not submitted its position regarding the automatic actuation of the SLCS for passive designs, the staff expects EPRI to provide design requirements for the passive designs that are consistent with those for the evolutionary designs. The passive ALWR vendors have indicated that their designs will comply with the Requirements Document for passive designs. The staff will evaluate their designs upon submittal of their final design approval/design certification application.

Because the policy aspects of this issue are resolved (pending review of the design application for the passive designs), the staff does not request any Commission action on this issue at this time. Should a policy question be raised as a result of the staff's review of this matter, the staff will inform the Commission at the earliest opportunity.

C. Mid-Loop Operation

In SECY-90-016, the staff stated that it was concerned that decay heat removal capability could be lost when a PWR is shut down for refueling or maintenance and drained to a reduced reactor coolant system (RCS) or "mid-loop" level. The staff recommended that the Commission approve its position that evolutionary PWR vendors propose design features to ensure high reliability of the shutdown decay heat removal system. In its April 26, 1990 letter, the Advisory Committee on Reactor Safeguards (ACRS) recommended that additional requirements be considered to resolve this issue. In its April 27, 1990 memorandum, the staff indicated that it would ensure that these recommendations were addressed in the evolutionary LWRs. In its June 26, 1990 SRM, the Commission approved the staff's position in conjunction with those of the ACRS.

The EPRI Requirements Document and evolutionary PWR designers have provided features to address the specific issue of mid-loop operation. The staff is in the process of evaluating the evolutionary plant submittals to ensure acceptable implementation of the Commission's directive.

However, the staff is concerned that the overall question regarding the vulnerability of the ALWRs during shutdown and low power operation has not been adequately evaluated by the vendors and EPRI. This issue has been discussed for evolutionary designs in a memorandum to the Commission dated September 5, 1990. The ALWR vendors and EPRI have been requested to assess shutdown and low power risk, identifying design specific vulnerabilities and weaknesses and documenting their consideration and incorporation of design features which minimize such vulnerabilities.

In its December 16, 1991 letter, EPRI submitted proposed changes to the Requirements Document to address this issue for the evolutionary designs. The staff has not yet received a response from the evolutionary vendors.

The passive ALWR vendors have indicated that their designs will comply with the Requirements Document for passive designs. The staff expects that EPRI will specify requirements for the passive designs similar to those of the evolutionary.

The staff concludes that passive plants must also have a reliable means of maintaining decay heat removal capability during all phases of shutdown activities, including refueling and maintenance, and will evaluate the adequacy of the passive Requirements Document and designs during its review. The staff does not consider this issue to be a policy matter, but rather an element of its normal review. Therefore, the staff does not request any Commission action on this issue at this time. Should a policy question be raised as a result of the staff's review of this matter, the staff will inform the Commission at the earliest opportunity.

D. Station Blackout

As discussed in SECY-90-016, the station blackout rule (10 CFR 50.63) allows utilities several design alternatives to ensure that an operating plant can safely shut down in the event that all ac power (offsite and onsite) is lost. The staff concluded that the preferred method of demonstrating compliance with 10 CFR 50.63 is through the installation of a spare (full capacity) alternate ac power source of diverse design that is consistent with the guidance in Regulatory Guide 1.155, and is capable of powering at least one complete set of normal shutdown loads. The staff recommended that the Commission approve imposition of an ac source for evolutionary ALWRs. In its June 26, 1990 SRM, the Commission approved the staff's position.

In addition to other design features to address the issue of station blackout, the EPRI Requirements Document for evolutionary designs and the evolutionary vendors have provided for a large capacity, diverse alternate ac power source (combustion turbine generator) with the capability to power one complete set of normal safe shutdown loads. The staff is in the process of evaluating their submittals to ensure acceptable implementation of the Commission's directive, but does not expect any related policy matters to result from its review.

Because the passive designs do not rely on active systems for safe shutdown following an event, EPRI and the passive plant designers do not believe safety-related diesel generators are necessary to address station blackout concerns. In addition, they believe an alternate ac power source should not be required. However, the staff concludes that the non-safety-related diesel generators may require some regulatory oversight. This issue is enveloped for the passive designs under the issue on regulatory treatment of non-safety systems (see paragraph III.A of this enclosure).

The staff is still evaluating this issue for the passive plant designs. It is identifying this issue in this paper to comply with the Commission's directives to present policy matters for Commission consideration at the earliest

opportunity. The staff's proposed resolution of this issue will be provided to the Commission later. Therefore, the staff does not request any Commission action on this issue at this time.

E. Fire Protection

As discussed in SECY-90-016, the staff recommended that current NRC guidance to resolve fire protection issues should be enhanced to minimize fire as a significant contributor to the likelihood of severe accidents for advanced plants. In its April 26, 1990 letter, the ACRS recommended that the staff should consider additional matters in its evaluation of the fire protection designs. In its June 26, 1990 SRM, the Commission approved the staff's position as supplemented by the staff's April 27, 1990 response to the ACRS.

The EPRI Requirements Document and the evolutionary ALWR designers have indicated that their fire protection designs are consistent with the staff's proposed enhancements. The staff is in the process of evaluating their submittals to ensure acceptable implementation of the Commission's directive, but does not expect any related policy matters to result from its review.

The staff expects that EPRI will specify requirements for the passive designs similar to those of the evolutionary. The passive ALWR vendors have indicated that their designs will comply with the Requirements Document for passive designs.

Although the staff is still evaluating this issue for the passive plant designs, it has not identified any unique features of the designs that would preclude the staff's conclusion that these designs should also be evaluated against the enhanced fire protection criteria. Therefore, the staff recommends that the Commission approve the staff's position that the passive plants should also be reviewed against the enhanced fire protection criteria approved in the Commission's June 26, 1990 SRM.

F. Intersystem Loss-of-Coolant Accident

As discussed in SECY-90-016, the staff recommended that future evolutionary ALWR designs reduce the possibility of a loss-of-coolant accident (LOCA) outside containment by designing (to the extent practicable) all systems and subsystems connected to the reactor coolant system (RCS) to withstand the full RCS pressure. However, for both the passive and evolutionary reactors, design stress allowables for intersystem LOCA conditions in low pressure piping systems and uniform criteria must yet be developed and approved by the staff. The staff further recommended that systems that have not been designed to withstand full RCS pressure should include

1. the capability for leak testing of the pressure isolation valves,

2. valve position indication that is available in the control room 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 attached low-pressure systems and both isolation valves are not closed.

In its June 26, 1990 SRM, the Commission approved the staff's position on intersystem LOCA provided that all elements of the low pressure system are considered (e.g., instrument lines, pump seals, heat exchanger tubes, and valve bonnets). The staff has been applying items 1 - 3 to systems that have not been designed to operate at full RCS pressure.

The EPRI Requirements Document and the evolutionary ALWR designers have indicated that their submittals are consistent with the approved resolution to this issue. The staff is in the process of evaluating their submittals to ensure acceptable implementation of the Commission's directive, but does not expect any related policy matters to result from its review.

The staff expects that EPRI will specify requirements for the passive designs similar to those of the evolutionary. The passive ALWR vendors have indicated that their designs will comply with the Requirements Document for passive designs.

Although the staff is still evaluating this issue for the passive plant designs, it has not identified any unique features of the designs that would preclude the staff's conclusion that these designs should also be evaluated against the staff's previous recommendation. Therefore, the staff recommends that the Commission approve the staff's position that the passive plants should also be reviewed against the position for intersystem LOCA as approved in the Commission's June 26, 1990 SRM.

G. Hydrogen Control

Containments are required to be designed for control of hydrogen generation following an accident. 10 CFR 52.47(a)(ii) requires all applicants for design certification to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f). 10 CFR 50.34(f) requires a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100 percent fuel-clad metal water reaction, and to assure that uniformly distributed hydrogen concentrations in the containment do not exceed 10 percent (by volume) or that the post-accident atmosphere will not support hydrogen combustion.

In SECY-90-016, the staff recommended that, because of the uncertainties in the phenomenological knowledge of hydrogen generation and combustion, evolutionary ALWRs should be designed, as a minimum, to

1. accommodate hydrogen equivalent to 100-percent metal-water reaction of the fuel cladding,

2. limit containment hydrogen concentration to no greater than 10 percent, and

3. provide containment-wide hydrogen control (e.g., igniters, inerting) for severe accidents.

The staff recommended that the Commission approve the staff's position that the requirements of 10 CFR 50.34(f)(2)(ix) remain unchanged for evolutionary ALWRs. In its June 26, 1990 SRM, the Commission approved the staff's position.

In its December 6, 1991 letter, EPRI stated that the EPRI Requirements Document for evolutionary designs will be modified to fully comply with the above positions. The ABWR design meets the requirements of 10 CFR 50.34(f)(2)(ix) by using, in conjunction with other systems, a nitrogen-inerted atmosphere within its containment. The System 80+ design proposes to be consistent with the recommendations of the ALWR Requirements Document resulting from staff review. The staff is in the process of evaluating these submittals to ensure acceptable implementation of the Commission's directive, but does not expect any related policy matters to result from its review.

The staff expects that EPRI will specify requirements for the passive designs similar to those of the evolutionary. The passive ALWR vendors have indicated that their designs will comply with the Requirements Document for passive designs.

Although the staff is still evaluating this issue for the passive plant designs, it has not identified any unique features of the designs that would preclude the staff's conclusion that these designs should also be evaluated against the staff's previous recommendation. Therefore, the staff recommends that the Commission approve the staff's position that the passive plants should also be designed, as a minimum, to

1. accommodate hydrogen equivalent to 100-percent metal-water reaction of the fuel cladding,

2. limit containment hydrogen concentration to no greater than 10 percent, and

3. provide containment-wide hydrogen control (e.g., igniters, inerting) for severe accidents.

H. Core Debris Coolability

In the unlikely event of a severe accident in which the core has melted through the reactor vessel, it is possible that containment integrity could be breached if the molten core is not sufficiently cooled. In addition, interactions between the core debris and concrete can generate large quantities of additional hydrogen and other non-condensible gases, which could contribute to eventual overpressure failure of the containment. Therefore, the staff concluded that plant designs should include features to enhance core debris coolability.

As discussed in SECY-90-016, the staff recommended that the Commission approve the general criteria that evolutionary ALWR designs

1. provide sufficient reactor cavity floor space to enhance debris spreading, and

2. provide for quenching debris in the reactor cavity.

In its June 26, 1990 SRM, the Commission approved the staff's position.

In addition, the staff indicated in SECY-90-016 that it was evaluating the level of protection afforded by covering the containment liner and other structural members with concrete. The staff concludes that it may be necessary to protect these structural components with concrete.

The EPRI Requirements Document and the evolutionary ALWR designs provide a number of design features that are intended to mitigate the effects of a molten core. Among other features, the evolutionary designs are proposing floor sizing criteria of 0.02 m^2/MWt and provisions to flood the lower drywell or reactor cavity. The staff does not support or dispute the EPRI floor sizing criteria of 0.02 m^2/MWt. Instead, it concludes that it is appropriate to review the specific vendor designs to determine how they have addressed the three items discussed above to provide an increased level of protection relative to core debris coolability. The staff concludes that the "core-on-the-floor" accident will not be considered as a new design basis accident. However, the staff expects the vendors to consider the effects of an unmitigated core-concrete interaction on the production of non-condensible gases, the release of additional fission products from the core-concrete interaction, and additional heat and hydrogen generation in the new designs. Further, the staff will evaluate how the vendors have addressed core debris interactions with water in the cavity to account for steam and hydrogen generation.

The three criteria discussed above are intended to ensure that the ALWR vendors provide measures to the extent practical to mitigate severe accidents while avoiding turning severe accidents into traditional design basis accidents. As the staff neither supports nor disputes particular floor sizing criteria, the vendors should ensure that the containment can withstand the pressure increases caused by core-concrete interactions. For the range of severe accidents of concern, the vendors should realistically estimate the amount of core-concrete interaction that will occur, and ensure that the containment will accommodate the resultant conditions for at least 24 hours. Where insufficient data exists to develop realistic estimates, the vendor may propose alternatives, such as additional tests or the use of other methodologies for determining the degree of core-concrete interaction. The ALWR vendors should also perform parametric studies to determine how sensitive the containment response is to variations in the amount of core-debris which is available to interact with the concrete. The staff concludes that incorporation of the mitigative measures to the extent practical and assurance of containment integrity for a 24 hour period will provide defense in depth as well as an appropriate degree of robustness in the containment design.

In its December 6, 1991 submittal, EPRI indicated that it will specify requirements for the passive designs similar to those of the evolutionary designs. The passive ALWR vendors have indicated that their designs will comply with the Requirements Document for passive designs.

Although the staff is still evaluating this issue for the passive plant designs, it has not identified any unique features of the designs that would preclude the staff's conclusions that these designs should also be evaluated against the staff's previous recommendation. Therefore, the staff recommends the Commission approve the staff's position that both the evolutionary and passive LWR designs

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, and

4. ensure that the containment can accommodate the pressure increases resulting from core-concrete interactions involving a range of scenarios which release core debris into the containment for 24 hours following the start of a severe accident.

I. High Pressure Core Melt Ejection

In SECY-90-016, the staff recommended that the Commission approve the staff's position that evolutionary ALWR designs should include a depressurization system and cavity design features to contain ejected core debris to reduce the potential for containment failure by direct containment heating (DCH). The staff is concerned that this phenomenon might occur from the ejection of molten core debris under high pressure from the reactor vessel resulting in wide dispersal of core debris, rapid oxidation, and extremely rapid addition of energy to the containment atmosphere.

In its June 26, 1990 SRM, the Commission approved the staff's position with the directive that the cavity design, as a mitigating feature, should not unduly interfere with operations, including refueling, maintenance, or surveillance activities. Examples of cavity design features that will decrease the amount of ejected core debris that reaches the upper containment include ledges or walls that would deflect core debris and an indirect path from the lower reactor cavity to the upper containment. The staff will review the LWR designs relative to the above criteria.

Notwithstanding the goal of providing a retentive cavity design, the inherent uncertainties in the industry's understanding of core dispersal phenomena following the failure of the reactor coolant system (RCS) lead to the recommendation that defense-in-depth be provided through a highly reliable RCS

depressurization system. Such a system would provide additional assurance that DCH would not occur by preventing reactor vessel failure at high pressure, and the rapid dispersal of core debris.

The EPRI Requirements Document and plant designers have provided features to address this issue for evolutionary designs. The staff is in the process of evaluating their submittals to ensure acceptable implementation of the Commission's directive. The staff's preliminary review of the passive ALWRs has also identified the importance of depressurization of the reactor coolant system (RCS) to the safe shutdown of the plant during transients or accidents. Depressurization of the RCS is crucial to the operation of the passive safety features that limit the likelihood of core damage, as well as to reducing the potential for containment failure by direct containment heating from the ejection of core debris at high pressure. Therefore, the staff has determined that the passive ALWR designs must include a highly reliable depressurization system. The staff will also be evaluating the capability of non-safety systems to provide coolant injection under high pressure conditions should the depressurization system fail (see paragraph III.A of this enclosure).

Although the staff is still evaluating this issue for the passive plant designs, it has not identified any unique features of the designs that would preclude the staff's conclusions that these designs should also be evaluated against the staff's previous recommendations. Therefore, the staff recommends the Commission approve the general criteria that the passive LWR designs

1. provide a reliable depressurization system, and

2. provide cavity design features to decrease the amount of ejected core debris that reaches the upper containment.

J. Containment Performance

As discussed in SECY-90-016, the staff recommended that the Commission approve the staff's position to use a conditional containment failure probability (CCFP) of 0.1 or a deterministic containment performance goal that offers comparable protection in the evaluation of evolutionary ALWRs. The staff concluded that the following general criterion for containment performance during a severe-accident challenge would be appropriate for the evolutionary ALWRs in place of a CCFP.

> The containment should maintain its role as a reliable leak tight barrier by ensuring that containment stresses do not exceed ASME service level C limits for a minimum period of 24 hours following the onset of core damage and that following this 24 hour period the containment should continue to provide a barrier against the uncontrolled release of fission products.

The containment performance goal was proposed to ensure that the containment will perform its function in the face of most credible severe accident challenges.

In its June 26, 1990 SRM, the Commission approved the use of a 0.1 CCFP as a basis for establishing regulatory guidance for the evolutionary LWRs. The Commission directed that this objective should not be imposed as a requirement, and that the use of the CCFP should not discourage accident prevention. The staff was directed to review suitable alternative, deterministically-established containment performance objectives providing comparable mitigation capability that may be submitted by the applicants. The Commission directed that any such alternatives be submitted to the Commission.

While the staff has identified, both in this paper and in SECY-90-016, the major challenges to the containment (e.g., hydrogen burns, corium interactions with water and containment structures), and the need to provide the means for mitigation of these challenges, the containment performance goal acts as a final check to ensure that the design (including its mitigation features) would be adequate if called upon to mitigate a severe accident. Although not explicitly identified in SECY-90-016, the staff will also evaluate the impact of molten fuel-coolant interaction and the resulting steam and hydrogen generation (and any dynamic forces due to ex-vessel fuel-coolant interactions) on the integrity of the containment, consistent with the containment performance goal. The intent of both the CCFP and the alternative deterministic performance criteria discussed above is to provide this final check, as well as defense-in-depth. The philosophy behind the use of the proposed deterministic goal is that adequate time must be provided for fission product decay before allowing a release from the containment to the environment. Since Service Level C is applicable only to metal containments, a comparable criterion is needed for the concrete containments, such as used in the ABWR and SBWR designs. The staff is evaluating options for this criterion.

In its letter dated December 16, 1991, EPRI stated that its overall containment performance requirements address the assurance of a robust containment for ALWRs capable of accommodating risk-significant severe accident loads without failure. The staff concludes that EPRI's position is consistent with the deterministic containment performance goal. EPRI has also identified a comprehensive list of containment challenges resulting from core damage sequences. Based on its containment performance studies for the evolutionary and passive plant designs, EPRI has concluded that the described design features limit the likelihood and magnitude of these challenges, and that they ensure the capability of the containment to accommodate all challenges which are potentially risk significant. The staff is in the process of evaluating the adequacy of the containment performance of the individual ALWR reactor designs to ensure that these and any other potential sequences that may be identified during the staff's review are adequately addressed. The staff will evaluate the criteria for the vendor's determination of the challenges to the containment. The evaluation of containment bypass sequences will be addressed on a vendor-specific basis during the staff's design reviews. The staff does not expect any related policy matters to result from its review of the evolutionary designs.

The staff expects that EPRI will specify requirements for the passive designs similar to those of the evolutionary. The passive ALWR vendors have indicated that their designs will comply with the Requirements Document for passive designs.

Although the staff is still evaluating this issue for the passive plant designs, the staff concludes that it is appropriate to apply the same containment performance goal to the passive designs as a basis for establishing regulatory guidance. Therefore, the staff recommends the Commission approve the staff's position to use a conditional containment failure probability (CCFP) of 0.1 or a deterministic containment performance goal that offers comparable protection in the evaluation of the passive ALWRs. The staff will consider any suitable alternative, deterministically-established containment performance objectives providing comparable mitigation capability. Applicants using the deterministic approach will be required to clearly define the challenges considered in this evaluation.

K. Dedicated Containment Vent Penetration

As discussed in SECY-90-016, the staff recommended that the Commission approve the use of an overpressure protection system that used a dedicated containment vent for the ABWR. This system is designed to avoid gross containment failure resulting from postulated slow rising overpressure scenarios that could result from postulated multiple safety system failures.

In its June 26, 1990 SRM, the Commission approved the staff's recommended use of the containment overpressure protection system for the ABWR, subject to a comprehensive regulatory review to weigh the "downside" risks with the mitigation benefits of the system. In addition, the Commission directed the staff to ensure that full capability to maintain control over the venting process is provided in the design.

In its December 6, 1991 letter, EPRI stated that the EPRI Requirements Document for evolutionary designs will be modified to require containment overpressure protection through the containment design (considering size and strength of the containment) or through the use of pressure relief. EPRI proposes that the need for a containment vent be determined on a design-specific basis. EPRI considers this to be a plant optimization subject. The ABWR includes a dedicated vent system in its containment design.

The staff is in the process of evaluating these submittals to ensure acceptable implementation of the Commission's directive, but does not expect any related policy matters to result from its review. The staff expects that EPRI will specify requirements for the passive designs similar to those of the evolutionary. The passive ALWR vendors have indicated that their designs will comply with the Requirements Document for passive designs.

Because of the stage of design development, the staff has insufficient information to determine whether a containment vent is necessary for passive plant designs at this time. The containment performance criteria proposed in

Section I.J of this enclosure will serve as the basis for the staff's review of containment integrity and the need for a containment vent. Therefore, the staff recommends that the Commission approve the staff's position that the need for a containment vent for the passive plant designs be evaluated on a design specific basis.

L. Equipment Survivability

As discussed in SECY-90-016, the staff recommended that the Commission approve the staff's position that features provided only for severe-accident protection need not be subject to the 10 CFR 50.49 environmental qualification requirements, 10 CFR Part 50, Appendix B quality assurance requirements, and 10 CFR Part 50, Appendix A redundancy/diversity requirements. The reasons for this judgement is that the staff does not believe that severe core damage accidents should be treated in the same manner that design basis accidents (DBAs) have traditionally been treated, due to the large differences in their likelihood of occurrence. However, SECY-90-016 further stated that mitigation features must be designed so there is reasonable assurance that they will operate in the severe-accident environment for which they are intended and over the time span for which they are needed. In instances where safety related equipment which is provided for design bases accidents is relied upon to cope with severe accident situations, there should also be a high confidence that this equipment will survive severe accident conditions for the period that it is needed to perform its intended function. During the review of the credible severe accident scenarios for specific ALWR designs, the equipment needed to perform mitigative functions, and the conditions under which the mitigative systems must function, will be identified. Equipment survivability expectations under severe accident conditions should include consideration of the circumstances of applicable initiating events (e.g., station blackout, earthquakes) and the environment (e.g., pressure, temperature, radiation) in which the equipment is relied upon to function. The required system performance criteria will be based on the results of these design-specific reviews. In its June 26, 1990 SRM, the Commission approved the staff's position.

In its letter dated May 6, 1991, the staff clarified its position that this criteria would be applied to those features provided only for severe accident mitigation.

The EPRI Requirements Document and the evolutionary ALWR designers have indicated that their submittals are consistent with this criteria. The staff is in the process of evaluating their submittals to ensure acceptable implementation of the Commission's directive, but does not expect any related policy matters to result from its review. The passive ALWR vendors have indicated that their designs will comply with the Requirements Document for passive designs.

Although the staff is still evaluating this issue for the passive plant designs, it has not identified any unique features of the designs that would preclude the staff's conclusion that these designs should also be evaluated

against this criteria. Therefore, the staff recommends that the Commission approve the staff's position that features provided only for severe-accident mitigation for the passive plant designs need not be subject to the 10 CFR 50.49 environmental qualification requirements, 10 CFR Part 50, Appendix B quality assurance requirements, and 10 CFR Part 50, Appendix A redundancy/diversity requirements. As discussed in SECY-90-016, the staff concludes that guidance such as that found in Appendices A and B of Regulatory Guide 1.155, "Station Blackout," is appropriate for equipment used to mitigate the consequences of severe accidents.

M. Elimination of Operating Bases Earthquake

In SECY-90-016, the staff discussed its proposal to decouple the operating basis earthquake (OBE) from the safe-shutdown earthquake (SSE). The regulations in Appendix A to 10 CFR Part 100 establish the OBE at one-half of the SSE. The staff stated that the OBE should not control the design of safety systems and was evaluating possible changes to the regulations that would reduce the magnitude of the OBE relative to the SSE. The staff recommended that the Commission approve the review approach to consider requests to decouple the OBE from the SSE on a design-specific basis for evolutionary designs. In its June 26, 1990 SRM, the Commission approved the staff's recommendation.

In a plant optimization subject, EPRI requested that the staff evaluate the elimination of the OBE altogether from design of systems, structures, and components in nuclear power plants. In its April 26, 1990 letter, the ACRS also recommended this approach. The NRC staff, in evaluating the decoupling of the OBE from the SSE, is also evaluating the possibility of redefining the OBE in order to satisfy its function without an explicit response analysis. This change would diminish the role of the OBE in design by establishing a level which, if it is exceeded, would require that the plant be shut down for inspection activities.

EPRI's position on seismic design is that it is unnecessary to perform two complete sets of seismic analyses -- one for the OBE and one for the SSE. The NRC staff agrees, in principle, with this position but finds that extant design practices for piping and structures do not result in designs that are significantly controlled by the OBE. As stated in SECY-90-016, certain interim measures, such as allowing higher damping values for piping analyses, have been already implemented to alleviate the situation of having the OBE significantly controlling the design.

The elimination of the OBE response analysis would require all current OBE design-related checks to be performed for a fraction of the SSE. The staff is currently developing various alternatives with the industry to revise the codes and standards when design-related checks are based on the OBE. For example, in piping design, the ASME Boiler and Pressure Vessel Code currently establishes rules for the evaluation of earthquake cycles on fatigue and relative seismic anchor motion effects that are based on the OBE. In addition, the NRC requirements for postulating the number and location of pipe

ruptures are also derived from the OBE. If the OBE is eliminated from design, then these loadings might need to be performed by using the SSE and establishing new appropriate allowable limits.

Therefore, the staff recommends that the Commission approve the staff's approach to eliminate the OBE from design of systems, structures, and components. Until the final rulemaking to Appendix A to 10 CFR Part 100 is completed, the elimination of the OBE from design of either the evolutionary or passive designs will require an exemption from the current regulations with acceptable supporting justification from the designer. The details of how current OBE-related design checks will be performed using the SSE will be resolved between the industry and the NRC staff through the appropriate code-related activities or supplemental regulatory guidance.

N. In-Service Testing of Pumps and Valves

As discussed in SECY-90-016, the staff recommended that the Commission approve the staff's position that the following provisions be applied to all safety related pumps and valves and not limited to ASME Code Class 1, 2, and 3 components.

1. Piping design should incorporate provisions for full flow testing (maximum design flow) of pumps, and check valves.

2. Designs should incorporate provisions to test motor operated valves under design basis differential pressure.

3. Check valve testing should incorporate the use of advanced non-intrusive techniques to address degradation and performance characteristics.

4. A program should be established to determine the frequency necessary for disassembly and inspection of pumps and valves to detect unacceptable degradation which cannot be detected through the use of advanced non-intrusive techniques.

The staff concluded that these requirements were necessary to provide an adequate level of assurance of operability.

In its June 26, 1990 SRM, the Commission approved the staff's position as supplemented in the staff's April 27, 1990 response to ACRS comments. The Commission further noted that due consideration should be given to the practicality of designing testing capability, particularly for large pumps and valves.

The EPRI Requirements Document and the evolutionary ALWR designers have indicated that their submittals are consistent with this criteria. The staff is in the process of evaluating their submittals to ensure acceptable implementation of the Commission's directive, but does not expect any related

policy matters to result from its review. The passive ALWR vendors have indicated that their designs will comply with the Requirements Document for passive designs.

Although the staff is still evaluating this issue for the passive plant designs, it has concluded that this is an issue for passive designs because the passive safety systems rely on the proper operation of this equipment (i.e., check valves, depressurization valves) to mitigate the effects of transients. This is further discussed under the issue regarding the definition of passive failures (see paragraph III.B of this enclosure). Therefore, the staff recommends that the Commission approve the staff's position that these requirements also be imposed on passive ALWRs.

II. Other Evolutionary and Passive Design Issues

The following is a discussion of the status and, where available, proposed resolution of issues that are applicable to both the evolutionary and passive ALWR designs.

A. Industry Codes and Standards

In SECY-91-273, the staff raised the concern that a number of design codes and industry standards dealing with new plant construction have been recently developed or modified, and that the NRC has not yet determined their acceptability. EPRI and the ALWR vendors are using codes and standards in their applications that the staff has not endorsed.

The staff recommends that the Commission approve the staff's position that, consistent with past practice, it use the newest codes and standards that have been endorsed by the NRC in its reviews of both the evolutionary and passive plant design applications. Unapproved revisions to codes and standards will be reviewed on a case-by-case basis.

B. Electrical Distribution

In SECY-91-078, the staff recommended that the Commission approve its position that an evolutionary plant design should include

1. an alternate power source to the non-safety loads unless the design can demonstrate that the design margins will result in transients for a loss of non-safety power event that are no more severe than those associated with the turbine-trip-only event in current existing plant designs, and

2. at least one offsite circuit to each redundant safety division supplied directly from one of the offsite power sources with no intervening non-safety buses in such a manner that the offsite source can power the safety buses upon a failure of any non-safety bus.

In its August 15, 1991 SRM, the Commission approved the staff's positions.

In its December 6, 1991 letter, EPRI stated that it was modifying the EPRI Requirements Document for evolutionary plants, and would be submitting its proposed resolution in a future response. The staff will evaluate EPRI's submittal and those of the evolutionary final design approval/design certification applicants to ensure acceptable implementation of the Commission's directive, but it does not expect any related policy matters to result from its review.

Because the passive designs do not rely on active systems for safe shutdown following an event, the staff has not determined the applicability of this issue to the passive designs. This issue is enveloped for the passive designs under the issue on regulatory treatment of non-safety systems (see paragraph III.B of this enclosure).

The staff is still evaluating this issue for the passive plant designs. It is identifying this issue in this paper to comply with the Commission's directives to present policy matters for Commission consideration at the earliest opportunity. Should this issue be identified as applicable to the passive plant designs, the staff will provide a proposed resolution to the Commission later. Therefore, the staff does not request any Commission action on this issue at this time.

C. Seismic Hazard Curves and Design Parameters

To assess the seismic risk associated with an ALWR design, EPRI has proposed the use of generic bounding seismic hazard curves for sites in the central and eastern United States. EPRI proposes that these curves be used in the seismic probabilistic risk assessment (PRA). The regulations do not require and the staff does not intend to require that a seismic PRA be performed to determine if a site is acceptable.

To assess the EPRI ALWR seismic hazard bounding curve for rock sites, the staff compared the EPRI curve to results derived by Lawrence Livermore National Laboratories (LLNL) using the historical earthquake method discussed in NUREG/CR-4885, "Seismic Hazard Characterization of the Eastern United States: Comparative Evaluation of the LLNL and EPRI Studies," 1987. The staff also compared the EPRI bounding curve to hazard curves generated by EPRI using the historical method for the Seabrook site (see letter dated October 17, 1991). The historical hazard curves below 0.1g reflect the past few hundred years of historical earthquake data. The historical hazard curves at higher accelerations are estimates based on the historical earthquake data. Both the LLNL and EPRI hazard curves, which were derived using the historical method, exceed the EPRI bounding curve at accelerations below about 0.1g. Because the EPRI bounding curve is exceeded at low peak accelerations by the results based on historical earthquake data, the staff also questions the adequacy of the EPRI bounding curve at higher peak accelerations.

Hazard curves generated for the Seabrook Station Probabilistic Safety Assessment (1983) by the licensee exceed the EPRI bounding hazard curve. The Seabrook SSE has a peak acceleration of 0.25g, whereas a higher SSE of 0.3g is proposed for ALWR sites. On the basis of the deterministic design basis of 0.3g, the EPRI-proposed criteria can be assumed to be suitable for the Seabrook site. However, based on the probabilistic assessment, the EPRI bounding hazard curve would underestimate the core damage frequency. Thus, the EPRI bounding hazard curve is non-conservative when compared to a licensee submittal.

The LLNL hazard curves are used in the staff's reviews of seismic hazard and are generally higher than the EPRI results for the same sites. Some LLNL hazard curves for sites in the Eastern United States (discussed in NUREG/CR-5250, "Seismic Hazard Characterization of 69 Nuclear Plant Sites East of the Rocky Mountains," 1989) exceed the EPRI bounding hazard curve.

During the staff's review of the ABWR PRA, PRA results using both the LLNL and EPRI hazard estimates were compared with results using the ABWR bounding seismic hazard curve. The ABWR bounding hazard curve was exceeded by the LLNL mean hazard curves for the Pilgrim, Seabrook, and Watts Bar sites. These three sites in the eastern United States were selected because of their relatively high seismic hazard. The staff used both the LLNL and EPRI seismic hazard estimates to quantify core damage frequency. The PRA using the LLNL hazard curves predicted much higher core damage frequencies than the PRA using the EPRI hazard curve. However, the ABWR design was determined to be capable of resisting earthquakes significantly larger than an SSE of 0.3g.

The evolutionary and passive ALWR designers have indicated that their applications will be consistent with the EPRI criteria. However, based on the staff's review of historical seismicity and the LLNL hazard estimates, the staff has concluded that the EPRI seismic hazard bounding curve is not sufficiently conservative. The staff is evaluating the seismicity and ground motion inputs used in the LLNL and EPRI studies to determine if the uncertainties in the curves can be reduced.

Based on the deterministic process used by the staff to judge the seismic capability of the GE and CE designs, the staff concludes that, with few exceptions, almost all areas east of the Rocky Mountains would be candidate sites for these designs. As part of the COL process, the applicant will have to demonstrate that the site-specific seismic parameters meet the certified design parameters to assure issue preclusion at the COL hearing. Should an actual site value exceed the design envelope in a certain area, a specific analysis will have to be performed to verify that the design is still acceptable for that site.

The discussion on seismic hazard curves provided in this section is for information only, and is provided to identify a complete list of issues applicable to the passive designs. If a policy question is identified as a result of its review, the staff will inform the Commission of the issue at the earliest opportunity. Therefore, the staff does not request any Commission action on this issue at this time.

D. Leak-Before-Break

General Design Criterion (GDC) 4 states, in part, that, "...dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping."

Under the broad scope revision to GDC 4 (52 FR 41288, October 27, 1987), the NRC allows the use of advanced technology to exclude from structural design consideration the dynamic effects of pipe ruptures in nuclear power plants provided it is demonstrated that the probability of pipe rupture is extremely low under conditions consistent with the design bases for the piping. Demonstration of low probability pipe rupture requires a deterministic fracture mechanics analysis that evaluates the stability of postulated small, through-wall flaws in piping and the ability to detect leakage through the flaws long before the flaw could grow to unstable sizes. The concept underlying such analyses is referred to as "leak-before-break (LBB)."

To date, the LBB approach has been approved by the NRC staff for currently operating and near-term operating licensed nuclear power plants based on a case-by-case review of plant-specific analyses. The NRC staff has approved the use of LBB for PWR primary coolant loop piping in all but five units in the United States. In addition, the pressurizer surge, accumulator, and residual heat removal piping have been approved for 11 units. In all cases, the LBB approvals have been granted for piping inside primary containment and for piping at least six inches nominal diameter. The piping includes both austenitic and carbon steel material. However, the LBB-approved carbon steel piping have all been clad with stainless steel material. To date, no BWRs have requested LBB approval.

EPRI and GE are proposing to adopt the LBB approach for ALWRs when certain details of the piping design, materials properties, and stress conditions are known. As discussed in SECY-89-013, the staff will evaluate the acceptability of the use of leak-before-break considerations in the ALWR designs when it can be justified. The staff has evaluated the EPRI and GE proposal for LBB application to ALWRs, as discussed below.

LBB Acceptance Criteria

The staff concludes that the analyses referred to in GDC-4 should be based on specific data, such as piping geometry, materials, and piping loads. The staff must review the LBB analyses for specific piping designs before the applicant can exclude the dynamic effects from the design basis. For ALWRs seeking design certification under 10 CFR Part 52, the analyses may be allowed to incorporate preliminary stress analysis results provided bounding limits (both upper- and lower-bound) are determined in order to establish assurance that adequate margins are available for leakage, loads, and flaw sizes.

A margin of 10 on leakage is required so that leakage from the postulated flaw size is assured of detection when the pipe is subjected to normal operational loads. A margin of 1.4 on loads is required to ensure that leakage-size flaws are stable at normal plus accident loads (e.g., safe-shutdown earthquake and safety-relief valve discharge loads). A factor of 2 between the leakage-size flaw and the critical-size flaw is required to ensure an adequate stability margin for the leakage-size flaw.

In addition, for ALWRs that seek approval of LBB during the design certification phase, certain information will be required for LBB analyses to establish through-wall flaw sizes and flaw stability. For through-wall flaw sizes, a lower-bound, normal-operational stress limit must be established for dead weight, pressure, and thermal loadings. The mean or best-estimate stress-strain curve should be used. For flaw stability, an upper-bound stress limit should be established for normal loadings plus safe shutdown earthquake and suppression pool hydrodynamic loadings. A lower-bound stress-strain curve for base metal should be used regardless of whether the weld or base metal is limiting. In addition, a lower-bound toughness for the weld metal should be used.

A deterministic fracture mechanics evaluation accounting for material toughness is required. Applicants may propose any fracture mechanics evaluation method for NRC staff review. However, the applicants will have to demonstrate the accuracy of the method by comparing it with other acceptable methods or with experimental data.

Using the above approach, an initial set of bounding values and a preliminary LBB analysis can be established for ALWRs during the design certification phase. These bounding values and preliminary analyses can be verified when as-built and as-procured information become available during the combined license phase. Prior to fuel-loading, verification of the preliminary LBB analyses should be completed and based on actual material properties and final, as-built piping analysis as part of inspection, tests, analyses, and acceptance criteria (ITAAC) associated with 10 CFR Part 52.

LBB Limitations

Because of the dependency of the LBB analyses to accurately predict the flaw stability, the LBB approach has established certain limitations for excluding piping that is likely to be susceptible to failure from various degradation mechanisms during service. A significant portion of the LBB review involves the evaluation of the susceptibility of the candidate piping in various degradation mechanisms to demonstrate that the candidate piping is not susceptible to failure from these degradation mechanisms. The operating history and measures to prevent or mitigate these mechanisms are reviewed by the NRC staff.

The LBB approach cannot be applied to piping that can fail in service from such effects as water hammer, creep, erosion, corrosion, erosion-corrosion, fatigue, thermal stratification, and environmental conditions. The rationale is that these degradation mechanisms challenge the assumptions in the LBB

acceptance criteria. For example, (1) water hammer may introduce excessive dynamic loads which are not accounted for in the LBB analyses, and (2) corrosion and fatigue may introduce flaws whose geometry may not be bounded by the postulated through-wall flaw in the LBB analyses. Adhering to the "defense-in-depth" principle, piping susceptible to failure from these potential degradation mechanisms is excluded from LBB applications. Alternatively, features to mitigate the possibility of certain degradation mechanisms may be proposed to ensure that LBB assumptions are not invalidated. For example, LBB might be considered for carbon steel piping for which the effects of erosion-corrosion have been eliminated through the use of high chromium steels with proven resistance to erosion-corrosion or through the use of carbon steel piping that is clad on the fluid surface with erosion-corrosion resistant materials.

A detailed discussion of the limitations and acceptance criteria for LBB used by the NRC staff is provided in NUREG-1061, Volume 3, "Evaluation of Potential for Pipe Breaks, Report of the U.S. Nuclear Regulatory Commission Piping Review Committee," dated November 1984.

Design Basis with LBB

The broad scope rule introduced an acknowledged inconsistency into the design basis by excluding the dynamic effects of postulated pipe ruptures while retaining non-mechanistic pipe rupture for containments, emergency core cooling systems (ECCSs), and environmental qualification (EQ) of safety-related electrical and mechanical equipment. The NRC staff subsequently clarified its intended treatment of the containment, ECCS, and EQ in the context of LBB application in a request for public comments on this issue that was published on April 6, 1988 (53 FR 11311).

Effects resulting from postulated pipe ruptures can be generally divided into local dynamic effects and global effects. Local dynamic effects of a pipe rupture are uniquely associated with that of a particular pipe rupture. These specific effects are not caused by any other source or even a postulated pipe rupture at a different location. Examples of local dynamic effects are pipe whip, jet impingement, missiles, local pressurization, pipe break reaction forces, and decompression waves in the intact portions of that piping or communicating piping. Global effects of a pipe rupture need not be associated with a particular pipe rupture. Similar effects can be caused by failures from sources such as pump seals, leaking valve packings, flanged connections, bellows, manways, rupture disks, and pipe ruptures of other piping. Examples of global effects are gross pressurization, temperatures, humidity, flooding, loss of fluid inventory, radiation, and chemical condition. For the ABWR, global effects also include suppression pool hydrodynamic loads (e.g., safety-relief valve discharges, pool-swell/fallback, condensation oscillation, and chugging loads). Although the suppression pool hydrodynamic loads caused by a main steam or feedwater pipe rupture might be excluded for the design of piping, equipment, and internal containment structures (other than those serving a containment function), the possibility of such dynamic effects being caused by a reactor internal pump ejection, failures of flanged connections,

and blowdowns from rupture disks or squib-actuated valves have not been addressed at this time. The option does exist to establish a postulated pipe break of a high-energy line smaller than the main steam of feedwater line break to envelope the possible global dynamic effects described above. This approach would be required to be submitted to the NRC staff for review and approval prior to its use. Until then, the use of a postulated pipe rupture of a main steam or feedwater line should be assumed for suppression-pool hydrodynamic loads.

The application of LBB technology eliminates the local dynamic effects of postulated pipe ruptures from the design basis. Because the global effects from the postulated pipe rupture provide a convenient and conservative design envelope, the NRC staff will continue to require the consideration of global effects for various aspects of the plant design, such as environmental qualification of equipment, design of containments, design of subcompartment enclosures, etc.

Recommendations

The elimination of dynamic effects of postulated high energy pipe ruptures from the design basis of ALWRs using advanced fracture mechanics analyses (leak-before-break approach) is permitted in the revised GDC 4 of Appendix A to 10 CFR Part 50. The limitations and acceptance criteria for LBB applications in ALWRs are the same as those established for currently operating nuclear power plants. Therefore, the staff recommends that the Commission approve the application of the LBB approach to both evolutionary and passive ALWRs seeking design certification under 10 CFR Part 52 when appropriate bounding limits are established during the design certification phase using preliminary analyses results and verified during the combined license phase by performing the appropriate ITAAC discussed herein. However, the specific details will need to be developed as the process is implemented during the first trial application.

E. Classification of Main Steamline of Boiling Water Reactors (BWRs)

Background

The main steamlines in boiling water reactor (BWR) plants contain dual quick-closing main steam isolation valves (MSIVs). These valves function to isolate the reactor system in the event of a break in a steamline outside of the primary containment, a design basis LOCA, or other events requiring containment isolation. Although the MSIVs are designed to provide a leak-tight barrier, it is recognized that some leakage through the valves will occur.

The current procedure for determining the acceptability of MSIV leakage involves calculating the dose in accordance with 10 CFR Part 100 based on a conservative assumption that the leakage limit allowed by the technical specification (normally 11.5 scfh per valve) is released directly into the environment. No credit is currently taken for the pressure integrity of the main steam piping and condenser.

Because of recurring problems with excessive leakage of MSIVs, Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants," recommended the installation of a supplemental leakage control system (LCS) to ensure that the isolation function of the MSIVs complies with the specified limits. Operating experience with the LCS has required substantial maintenance and worker exposure. Additionally, the NRC has generic concerns with the effectiveness of the LCS to perform its intended function under conditions of high MSIV leakage (Generic Issue C-8, Main Steam Line Valve Leakage Control Systems).

These concerns led EPRI to propose an alternative approach to ensure that doses associated with MSIV leakage would be acceptably low. EPRI identified this issue as a plant optimization subject. The resolution proposed by EPRI would eliminate the safety-related leakage control system, allow higher leakage limits through the MSIVs, and use an alternate MSIV leakage treatment method that takes credit for the large surface volume in the main steam piping and condenser hotwell to plate-out the fission products following core damage. In this way, the main steam piping (and by-pass line) and the condenser are used to mitigate the consequences of an accident. Appendix A to 10 CFR Part 100 requires that structures, systems, and components necessary to assure the capability to mitigate the consequences of accidents remain functional during and after a safe shutdown earthquake (SSE). These components are classified as safety-related and Seismic Category I. In addition, Appendix B to 10 CFR Part 50 establishes quality assurance requirements for safety-related, seismic Category I systems, structures, and components.

Discussion

Section 3.2.2 of the SRP recommends that the main steamline from the outermost isolation valve up to, but not including, the turbine stop valve including branch lines up to the first valve, be classified as Quality Group B (Safety Class 2). Regulatory Guide 1.29 designates such piping as Seismic Category I. The staff concludes that the main steam piping from the outermost isolation valve up to the seismic interface restraint and branch lines up to the first closed valve should conform to Appendix A of Section 3.2.2 of the SRP and Regulatory Guide 1.29. The main steamline from the seismic interface restraint up to but not including the turbine stop valve should be classified as Quality Group B but may be classified as non-seismic Category I. However, all pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 are applicable to this portion of the main steamline from the seismic interface restraint to the turbine stop valve. These requirements are needed to assure that the quality of the piping material is commensurate with its importance to safety during both operational and accident conditions.

The seismic interface restraint must provide a structural barrier between the Seismic Category I portion of the main steamline in the reactor building and the non-seismic Category I portions of the main steamline in the turbine building. The seismic interface restraint must be located inside the Seismic Category I building. The classification of the main steamline in the turbine building as non-seismic Category I is needed for consistency with the classification of the turbine building. On this basis, the quality and safety

requirements imposed on the main steamline from the outermost isolation valve up to the turbine stop valve are equivalent to the staff guidelines in Appendix A to Section 3.2.2 of the SRP and Regulatory Guide 1.29.

To ensure the integrity of the by-pass piping from the first valve to the main condenser hotwell, the staff and EPRI both agree that preventing gross structural failure of the piping and hotwell would provide assurance that leakage from the MSIVs following a design basis accident would not exceed the 10 CFR Part 100 guideline. The issue remaining is the classification of the main steam by-pass piping between the first normally-closed valve and the condenser hotwell as well as the hotwell itself. The staff proposes that the main steam by-pass line from the first valve up to the condenser inlet, and the piping between the turbine stop valve and the turbine inlet should not be classified as safety-related nor as seismic Category I, but should be analyzed using a dynamic seismic analysis to demonstrate its structural integrity under SSE loading conditions.

The staff proposes that the condenser be seismically analyzed to ensure that it is capable of maintaining its structural integrity during and after the SSE. The dose analysis considers that the condenser is open to atmosphere. Thus, it is only necessary to ensure no gross structural failure of the condenser. Similarly, it is only necessary to ensure that failure of non-safety related systems, structures, and components due to a seismic event do not cause failure of the main steam piping, by-pass line, or the condenser. The staff is developing design acceptance criteria for seismic analyses of piping systems which would become part of the inspections, tests, analyses, and acceptance criteria (ITAAC) for the COL holder to perform (see Section II.L of this enclosure). The staff proposes a similar approach for the above non-safety related systems, structures, and components.

Recommendations

The staff concludes that the above-described approach for both evolutionary and passive ALWRs to resolve the BWR main steam line issue provides reasonable assurance that the main steam piping from the outermost isolation valve up to the turbine stop valve, the main steam by-pass line up to the condenser, and the main condenser will retain their pressure and structural integrity during and following a safe shutdown earthquake. The proposed credit for non-safety related components (e.g., main steam by-pass piping and condenser) to mitigate the consequences of a design basis accident might require an exemption from the regulations. The staff recommends that the Commission approve the above-described approach for the resolution of the main steamline classification for both the evolutionary and passive ALWRs.

F. Tornado Design Basis

The current NRC regulatory position with regard to design basis tornados is contained in two documents written in 1974, WASH-1300, "Technical Basis for Interim Regional Tornado Criteria," and Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants." WASH-1300 states that the probability of

occurrence of a tornado that exceeds the Design Basis Tornado (DBT) should be on the order of 10E-7 per year per nuclear power plant and the regulatory guide delineates the maximum wind speeds of 240 to 360 per hour depending on the regions.

As a result of EPRI's earlier efforts on this EPRI-proposed plant optimization subject, the regulatory positions in Regulatory Guide 1.76 were reevaluated by an NRC contractor using the considerable quantity of tornado data which is now available but was not when the regulatory guide was developed. The contractor completed its reevaluation as discussed in NUREG/CR-4661 (PNL-9697), "Tornado Climatology of the Contiguous United States," dated May 1986. At the heart of this study is the tornado data tape prepared by the National Severe Storm Forecast Center (NSSFC) with 30 years of data, 1954 through 1983. This data tape contains the data for the approximately 30,000 tornados that occurred during the period.

The contractor found that the tornado strike probabilities range from near 10E-7 per year for much of the western United States to about 10E-3 per year in the central United States. Based on discussions between the contractor and the staff, wind speed values associated with a tornado having a mean recurrence interval of 10E-7 per year were estimated to be about 200 mph for the United States west of the Rocky Mountains and to be 300 mph for the United States east of the Rocky Mountains.

In its December 6, 1991 letter, EPRI proposed that a maximum tornado wind speed of 300 mph and a tornado recurrence interval of 10E-7 per year tornado strike probability be used for the design basis tornado to be used in the design of the evolutionary ALWRs. During a January 30, 1992 meeting with the staff, EPRI indicated that it would delete the reference to the tornado recurrence interval from the Requirements Document. The evolutionary and passive ALWR designers have indicated that their applications will be consistent with the EPRI Requirements Document.

The tornado design basis requirements have been used in establishing structural requirements (e.g., minimum concrete wall thicknesses) for the protection of nuclear plant safety-related structures, systems, and components against the effects not covered explicitly in review guidance such as Regulatory Guides or the SRP. Specifically, some aviation (general aviation light aircraft) crashes, nearby explosions, and explosion debris or missiles have been reviewed and evaluated routinely by the staff by taking into account the existence of the tornado protection requirements. The staff's acceptance of EPRI's proposal will also necessitate a concurrent review and evaluation of their effect on the protection criteria for some external impact hazards, such as general aviation or nearby explosions. Therefore, external impact hazards will be reviewed on a site-specific basis.

Based on the updated tornado data and the analysis provided in NUREG/CR-4661, the staff concludes that it is acceptable to reduce the tornado design basis wind speeds to 200 mph for the United States west of the Rocky Mountains, and to 300 mph for the United States east of the Rocky Mountains. Therefore, the staff recommends that the Commission approve the staff's position that a

maximum tornado wind speed of 300 mph be used for the design basis tornado to be used in the design of evolutionary and passive ALWR designs. As part of the COL process, the applicant will have to demonstrate that a design capable of withstanding a 300 mph tornado will also be sufficient to withstand other site hazards.

The staff expects that the use of this criteria will not preclude siting the ALWR plant designs on most sites in the United States. However, should an actual site hazard exceed the design envelope in a certain area, a specific analysis will have to be performed to verify that the design is still acceptable for that site.

G. Containment Bypass

The phenomenon of containment bypass is associated with the failure of the containment system to channel fission product releases through the suppression pool, or the failure of passive containment cooling system heat exchanger tubes in the large pools of water outside the containment. The fundamental characteristic of a BWR pressure-suppression containment is that steam that is released from the reactor coolant system will be condensed and scrubbed of radionuclides in a pool of water (the suppression pool) and the pressure rise in the containment will thereby be limited. This is accomplished by directing the steam from the reactor coolant system to the suppression pool through a vent system, but leakage paths could exist in this pathway between the drywell and the wetwell airspace that could allow steam to bypass the suppression pool and might over-pressurize the containment. Potential sources of steam bypass include leakage through the vacuum relief valves, cracking of the drywell structure, and penetrations through the drywell structure. In addition, a containment design which uses an external heat exchanger carries with it the potential for containment bypass from a leak in the heat exchanger. High temperatures associated with severe accidents or core debris carried from the reactor vessel could threaten the integrity of the heat exchanger tubes, and therefore provide a pathway for the release of fission products. Bypass of internal suppression pools could lead to overpressurization of the containment, and threaten its integrity. The staff believes that vendors should make reasonable efforts to minimize the possibility of bypass leakage, and should also allow for a certain amount of leakage in the containment design.

The provision of containment sprays in the drywell and/or wetwell would also reduce the impact of suppression pool bypass leakage on containment performance. These systems spray water into the containment and lower its temperature and pressure. They also scrub the containment atmosphere of fission products, and mitigate the effects of bypass on fission product distribution. In view of the contribution they can make to accident management, the staff is evaluating the need for containment spray systems for all ALWRs.

The discussion on steam bypass of the suppression pool is for information only and is provided to identify a complete list of issues applicable to the advanced designs. If a policy question is identified as a result of the

staffs' review, the Commission will be informed of the issue at the earliest opportunity. Therefore, the staff does not request any Commission action on this issue at this time.

H. Containment Leak Rate Testing

EPRI proposed that the maximum interval between Type C leakage rate tests should be 30 months rather that the 24 month maximum interval currently required in Appendix J to 10 CFR Part 50 for both evolutionary and passive plant designs. This plant optimization subject was proposed by EPRI to allow some margin between the nominal 24 month refueling interval and the Type C test interval to ensure that plant shutdowns will not be required solely to perform Type C tests. Other issues (air lock testing, Type C leak testing methods) have also been raised, but have not been raised to the Commission as a policy question.

In parallel with the staff's review of this issue on the ALWRs, the staff has developed proposed changes to Appendix J of 10 CFR Part 50 for all reactors, which were sent to the Commission in SECY-91-348. This document proposes modification to the regulation that would allow the increased interval as well as address other issues raised by EPRI. The staff's justification for these modifications is provided in SECY-91-348.

In addition, the staff has extended the time interval for performing Type C leakage rate testing on currently operating plants on a case-by-case basis (by as much as 1 year).

Therefore, the staff recommends that the Commission approve the staff's position that, until the rule change proceedings for Appendix J of 10 CFR 50 are completed, the maximum interval between Type C leakage rate tests for both evolutionary and passive plant designs be 30 months rather than the 24 month maximum interval currently required in Appendix J to 10 CFR Part 50.

I. Post-Accident Sampling System

10 CFR 50.34(f)(2)(viii) requires the designer to provide the

> ...capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain TID-14844 source term radioactive materials without radiation exposures to any individual exceeding 5 rem to the whole body or 75 rem to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage(e.g., noble gases, iodines and cesiums, and non-volatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations.

Regulatory Guide 1.97 and NUREG-0737, "Clarification of TMI Action Plan Requirements," provide guidance regarding the design of the post-accident sampling system (PASS) used to implement this regulation.

EPRI has proposed deviation from the design requirements for the PASS in several areas, as discussed below. EPRI has identified this issue as an optimization subject.

Elimination of the Hydrogen Analysis of Containment Atmosphere Samples

EPRI has stated that the hydrogen analysis of the containment atmosphere can be accomplished by the safety-grade containment hydrogen monitor required by 10 CFR 50.34(f)(2)(xvii) and II.F.1 of NUREG-0737. The staff concludes that adequate capability for monitoring post-accident hydrogen is provided by the safety-grade instrumentation, and that this is acceptable justification for an ALWR vendor to use in requesting this deviation. Because this exemption has been granted previously on currently operating plants, the staff does not consider this request to be a policy matter.

Elimination of Dissolved Gas and Chloride Analyses of Reactor Coolant Samples

EPRI considers the analyses of the reactor coolant for dissolved gas and chloride to be unnecessary because gases accumulated in the reactor vessel (mainly hydrogen) will be removed by venting, and corrosion due to the presence of chloride and oxygen will be minimized by prompt depressurization and cooling. Additionally, the amount of dissolved hydrogen in the reactor coolant can be determined based upon the hydrogen concentration measured in the containment atmosphere. 10 CFR 50.34(f)(2)(vii) and Item II.B.3 of NUREG-0737 specify that the PASS should have the capability to analyze dissolved hydrogen, oxygen, and chloride. This requirement was formulated before reactor vessels were required to have vents. With vented reactor vessels, the information on hydrogen concentration became less important.

However, the staff concludes that even with vented reactor vessels, there are some postulated accident sequences in which the reactor coolant system (RCS) is intact at reduced pressure, and heat is being removed by subcooled decay heat removal, e.g., similar to the TMI-2 accident. For these cases, it will not be possible to evaluate concentrations of the dissolved gases in the reactor coolant from their concentrations measured in the containment. The information on the amounts of dissolved hydrogen, chloride and oxygen in the reactor coolant is an important factor in evaluating post-accident conditions existing in the reactor vessel. The presence of hydrogen can affect flow of coolant in the core, and chloride and oxygen can cause corrosion of reactor components that may still be significant even at a low reactor coolant temperature. Therefore, the requirement for PASS sampling of coolant should not be eliminated. However, the staff agrees that sampling 24 hours after the end of power operation would be adequate to assure long term decay heat removal.

Therefore, the staff recommends that the Commission approve the staff's position that the capability to analyze dissolved hydrogen, oxygen, and chloride in accordance with the requirements of 10 CFR 50.34(f)(2)(viii) and II.B.3 of NUREG-0737 be required of the post-accident sampling system for the evolutionary and passive ALWRs. However, the time for taking these samples can be extended to 24 hours after the accident.

Relaxation in the Time Requirement for Sampling Activity Measurements

EPRI states that if boron solution has been added to permit plant shutdown, reactor water samples can be taken for boron analyses starting eight hours after the end of power operation. EPRI states the samples for activity measurements will not be required for 24 hours after the accident.

Item II.B.3 of NUREG-0737 specifies that the PASS should have the capability to obtain coolant and containment atmosphere sampling results within three hours from the time after the accident. The purpose of this requirement is to ensure that the capability exists for sampling in cases when samples are needed while the accident is still in progress. EPRI has committed that the neutron flux monitoring instrumentation will comply with the Category 1 criteria of Regulatory Guide 1.97. Therefore, this instrumentation will have fully qualified, redundant channels that monitor over the power range of 10E-6 percent to full power. Based on this commitment, the staff concurs with EPRI that boron sampling will not be required for the first 8 hours after an accident. Samples for activity measurements are required for evaluating the condition of the core. During the accident management phase, this information will be provided by the containment high range area radiation monitor, the containment hydrogen monitor, the reactor vessel water level indicator for BWRs, and the core exit thermocouples for PWRs. These data will be sufficient to meet the needs of the plant operators for the first 24 hours after an accident. The need for PASS activity measurements will occur during the accident recovery phase when the degree of core damage and general plant contamination will have to be evaluated. Based on this justification, the staff concludes that the requested extension of time for sampling activity measurements 24 hours after an accident is acceptable.

Therefore, the staff recommends that the Commission approve the staff's position that the requirements for sampling reactor coolant for boron and for activity measurements using the post-accident sampling system in the evolutionary and passive ALWRs deviate from the requirements of Item II.B.3 of NUREG-0737. The modified requirement would require the capability to take samples for boron and for activity measurements 8 hours and 24 hours, respectively, after the end of power operation.

J. Level of Detail

In its February 15, 1991 SRM on SECY-90-377, the Commission provided guidance regarding the level of detail of information required to determine the adequacy of design certification applications under 10 CFR Part 52. Although the level of detail issue is applicable to all design certification applications, the staff has been reviewing the ABWR as the lead plant in resolving this issue.

The staff identified several areas in the ABWR application where additional information is needed in order to resolve its safety concerns. The design detail resulting from the resolution of all of the staff's safety concerns will constitute the level of detail needed to support design certification in

accordance with the SRM on SECY-90-377. The level of design detail ultimately required by the staff could affect the schedules for all of the standard plant applications that the NRC has received to date.

In a meeting with GE, senior NRC managers and the vendor discussed certain areas of the review for which the designer has not provided final design details. These areas include piping design, leak-before-break, control room design, radiation protection, and advanced instrumentation and controls. The staff and GE agreed to pursue the development of design acceptance criteria with associated NRC "check points" as a substitute for detailed design information for a few limited areas of the design. To accomplish this, the NRC safety determination at design certification would be based on acceptance criteria that are general in nature, but measurable and observable. The "check points" would serve as milestones to confirm compliance with system requirements and the acceptance criteria. These issues would be documented in the Safety Analysis Report and the inspections, tests, analyses, and acceptance criteria (ITAAC), as appropriate.

The staff concludes that the level of detail issue is applicable to all design certification applications, but expects it to be resolved in the context of the ABWR review. The discussion provided in this section is for information only, and is provided to identify a complete list of issues applicable to the passive designs. The staff's proposed resolution of this issue will be provided to the Commission later. Therefore, the staff does not request any Commission action on this issue at this time.

K. Prototyping

SECY-91-074 discussed the process that the staff will use for determining the need for a prototype or other demonstration facility for the advanced reactor designs. The staff stated it will follow the procedure outlined in the paper to determine the various types of testing, up to and including a prototype facility, that may be needed to demonstrate that the advanced reactor designs are sufficiently mature to be certified.

Because the need for prototype testing is a design specific issue, this issue cannot be resolved during the EPRI review. The staff is in the process of evaluating the submittals of the evolutionary ALWRs to determine the need for prototype testing, but has not identified any areas that may require such testing yet.

As discussed in SECY-91-273, the necessity for separate effects and scaled integral testing for passive designs is under consideration.

The discussion on prototyping provided in this section is for information only, and is provided to identify a complete list of issues applicable to the passive designs. If a policy question is identified as a result of its review, the staff will inform the Commission of the issue at the earliest opportunity. Therefore, the staff does not request any Commission action on this issue at this time.

L. Inspections, Test, Analyses, and Acceptance Criteria (ITAAC)

10 CFR 52.79(c) requires that

> The application for a combined license must include the proposed test, inspections, and analyses which the licensee shall perform and the acceptance criteria therefor which are necessary and sufficient to provide reasonable assurance that, if the tests, inspections, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license. Where the application references a certified standard design, the tests, inspection, analyses, and acceptance criteria contained in the certified design must apply to those portions of the facility design which are covered by the design certification.

SECY-91-178 provided the staff's recommendations concerning the form and content of the inspections, test, analyses, and acceptance criteria (ITAAC) for a design certification rule and a combined license as required by 10 CFR Part 52. SECY-91-210 requested Commission guidance on an industry proposal that would allow the NRC staff to issue standardized plant final design approvals (FDAs) before the final staff approval of the proposed ITAAC. In its September 24, 1991 SRM, the Commission provided guidance regarding development of ITAACs for final design approval/design certification applications.

GE has been identified by the Nuclear Management and Resources Council (NUMARC) as the industry lead for developing ITAAC submittals. The staff will review the ITAAC submittals when they are received from the vendors, currently expected from GE in January 1992 for the ABWR. CE plans to submit the ITAAC for the System 80+ following GE's submittal, currently scheduled for May 1992.

The staff concludes that the ITAAC issue is applicable to all final design approval/design certification applications, but expects it to be resolved in the context of the ABWR review. The discussion provided in this section is for information only, and is provided to identify a complete list of issues applicable to the passive designs. The staff's response to the September 24, 1991 SRM is scheduled for completion in March 1992. The staff will continue to interact with the industry on this matter. Should additional policy questions be raised as a result of the staff's review, the staff will inform the Commission at the earliest opportunity. Therefore, the staff does not request any new Commission action on this issue at this time.

M. Reliability Assurance Plan (RAP)

In SECY-89-013, the staff stated that a program to ensure that the design reliability of safety significant systems, structures, and components is maintained over the life of a plant, referred to as the reliability assurance program (RAP), would be required for design certification. The ALWR vendors and EPRI were informed that the staff was considering this matter in November 1988.

The need for a safety-oriented reliability effort for the nuclear industry was initially expressed by the NRC in Section II.C.4 of NUREG-0600, "NRC Action Plan Developed as a Result of the TMI-2 Accident," dated August 1980. Initial research in this area focused on enhancing the reliability of safety systems and their supporting auxiliary systems. Follow-on research examined the feasibility and cost effectiveness of applying an aerospace-proven reliability concept to commercial nuclear reactors as a regulatory option to address the issue concerning anticipated transients without scram (ATWS). Work on this project continued until March 1985 when the research on operational safety reliability was consolidated into the research efforts on risk-based technical specifications.

Other research was performed for the NRC in support of operating and new reactor licensing review activities. NASA performed research that involved the application of NASA's Systems Assurance Analysis, which addressed reliability, safety, and quality concerns, to selected nuclear plant systems. In addition, as part of the license application review process for the Clinch River Breeder Reactor Program (CRBR), the NRC reviewed the CRBR RAP that was proposed by the Department of Energy.

The staff is working on the development of a detailed guidance document for the development of a reliability assurance program for ALWRs as well as reviewing the ALWR vendor's final design approval/design certification applications and the EPRI Requirements Document for evolutionary and passive plant designs. The staff views the RAP for ALWRs as a program that exists at two distinct levels: the first level applies to vendor submittals for final design approval/design certification; and the second level is applicable to a referencing applicant for a construction and operating license. The first level involves a top-level program that defines the scope, conceptual framework, and essential elements of an effective RAP. The second level fully develops and implements the program based on the plant-specific design information.

The staff is working with EPRI and the ALWR vendors on the development of the first level RAP for ALWRs. Should policy questions be raised as a result of the staff's review, the staff will inform the Commission at the earliest opportunity. The discussion provided in this section is for information only, and is provided to identify a complete list of significant issues applicable to the ALWR designs. In accordance with the Commission's November 12, 1991 SRM, the staff will include a discussion of issues concerning the reliability assurance program in its periodic updates on the advanced reactor program. Therefore, the staff does not request any Commission action on this issue at this time.

N. Site-Specific Probabilistic Risk Assessments (PRAs)

10 CFR 52.47 requires all applicants seeking standard design certification to provide a probabilistic risk assessment (PRA). However, details of the specific site characteristics where a plant would be sited are not required until the combined operating license (COL) licensing stage. Rather than

provide a PRA that envelopes all potential sites for all hazards (e.g., volcanism or external flooding on a river site), the staff has sought enveloping analyses for seismic events and tornadoes from the final design approval/design certification applicants, and concludes that COL applicants should provide site-specific PRA information that deals with other potential hazards specific to the particular site. Therefore, the staff recommends that the Commission approve the staff's position that site-specific PRA information be submitted at the COL stage that addresses applicable site-specific hazards such as river flooding, storm surge, tsunami, vulcanism, and hurricanes, and that enveloping analyses for seismic events and tornadoes be required from the final design approval/design certification applicants.

O. Severe Accident Mitigation Design Alternatives (SAMDAs)

As discussed in SECY-91-229, the National Environmental Policy Act (NEPA), Section 102(C)(iii) requires, in part, that

> ...all agencies of the Federal Government shall...(C) include in every recommendation or report on proposals for legislation and other major Federal actions significantly affecting the quality of the human environment, a detailed statement by the responsible official on...(iii) alternatives to the proposed action.

The U.S. Court of Appeals, in Limerick Ecology Action v. NRC, 869 F.2d 719 (3rd Cir. 1989), effectively required the NRC to include consideration of certain severe accident mitigation design alternatives (SAMDAs) in the environmental impact review performed as part of the operating license application for the Limerick Generation Station. The staff has concluded that a NEPA evaluation in the form of an environmental impact statement that considered SAMDAs would be an essential element of an application for a combined license under Subpart C of 10 CFR Part 52, for those applications that reference a design certified under Subpart B.

Therefore, in SECY-91-229, the staff recommended that the Commission approve the staff's recommendations to

1. address SAMDAs for certified designs in a single rulemaking,

2. approve the staff's approach for considering the costs and benefits of the review of SAMDAs for standard plant design certification, and

3. approve the staff's proposal to advise applicants for a final design approval/design certification that they must assess SAMDAs and provide the rationale supporting their decisions.

In its October 25, 1991 SRM, the Commission approved the staff's recommendations and requested they be kept informed on the staff's progress in its evaluation of the SAMDAs for final design approval/design certification applications.

Consistent with the third recommendation of SECY-91-229, the staff requested the ALWR vendors to assess SAMDAs for their designs and provide their rationale for determining whether the SAMDAs would improve the safety of those designs. The staff will evaluate the applicants' responses to these inquiries upon receipt of their submittals.

The staff concludes that the SAMDAs issue is applicable to all final design approval/design certification applications, but expects it to be resolved in the context of the ABWR and System 80+ reviews. The discussion provided in this section is for information only, and is provided to identify a complete list of issues applicable to the passive designs. The staff will continue to interact with the industry on this matter. Should additional policy questions be raised as a result of the staff's review, the staff will inform the Commission at the earliest opportunity. Therefore, the staff does not request any Commission action on this issue at this time.

P. Generic Rulemaking Related to Design Certification

SECY-91-262 provided the Commission with the staff's recommendations regarding generic rulemaking related to design certification. The staff recommended that the Commission

1. approve the staff's proposal to proceed with the design-specific rulemakings through individual design certifications to resolve selected technical and severe accident issues for the ABWR and System 80+ designs, and

2. note the staff's intent to proceed with generic rulemaking where appropriate for evolutionary and passive designs, as information becomes available from ongoing efforts on these issues, independent of the design review and certification processes.

The staff has not yet received Commission guidance on SECY-91-262.

As discussed in SECY-91-262, the staff concludes that consideration of generic rulemaking in lieu of design-specific rulemaking is applicable to all final design approval/design certification applications. However, the design of the passive plants is not sufficiently developed at this time for the staff to determine whether generic rulemaking should be initiated for the passive plant designs. Certain generic rulemaking activities related to the evaluations of source terms during postulated severe accidents are ongoing and the results of these rulemakings may be used during design certification of the passive reactor designs. Currently, this work is focused on updating 10 CFR Part 100 to separate siting criteria from reactor design criteria. The staff plans to incorporate the revised source term criteria in 10 CFR Part 50, and, in addition, to later consider the incorporation of generic severe accident criteria. The discussion provided in this section is for information only, and is provided to identify a complete list of issues applicable to the

passive designs. Should additional policy questions be raised as a result of the staff's review, the staff will inform the Commission at the earliest opportunity. Therefore, the staff does not request any new Commission action on this issue at this time.

III. Passive Design Issues Only

The following is a discussion of the issues that the staff has identified while reviewing the conceptual design information from the passive plant vendors in conjunction with the EPRI Requirements Document for passive designs. These issues are unique to the passive plant designs.

A. Regulatory Treatment of Non-Safety Systems

In contrast to both the current generation of light water reactors and the evolutionary ALWRs, the passive ALWR designs rely on safety systems which use the driving forces of buoyancy, gravity, and stored energy sources. These passive systems supply safety injection water, provide core and containment cooling, and perform other functions. There are no pumps in these passive safety systems, and all valves are powered by dc electric power from batteries, are air-operated, or use check valves actuated by the pressure differential across the valve. EPRI and the passive reactor vendors propose that these designs do not include safety-grade ac electric power.

The passive ALWR designs also include non-safety-grade active systems to provide defense-in-depth capabilities for reactor coolant makeup and decay heat removal. These systems serve as the first line of defense in the event of transients or plant upsets to reduce challenges to the passive systems. These active systems include: (1) the chemical and volume control system and control rod drive system, which provide reactor coolant makeup for the AP-600 and 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 which are needed to support these functions, including non-safety standby diesel generators. In addition, the passive ALWR designs also include non-safety-grade active systems, such as the control room heating, ventilating, and air conditioning system, for mitigation of the radiological consequences of an accident. Many of these systems traditionally have been safety-grade systems, but in the passive plants, they are not designed to meet safety-grade criteria, and credit is not taken for them in the Chapter 15 licensing design basis accident analyses. In SECY 90-406, the staff identified the role of these non-safety systems in the passive designs as an emerging technical issue.

Associated with the new, passive design approach, the licensing design basis analysis relies 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 of lack of a proven operational performance history. For example, there are uncertainties about the performance of check valves in the passive safety systems which 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 (ECCS) systems are more likely to overcome stuck valves). These uncertainties enhance the importance of the active non-safety systems in providing the defense-in-depth to prevent and mitigate accidents and core damage. Therefore, the staff's review of the passive designs requires a review of not only the passive safety systems, but also the functional capability and availability of the active non-safety systems to provide significant defense-in-depth and accident and core damage prevention capability.

For those active systems that perform defense-in-depth functions, the EPRI Requirements Document for passive designs 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 EPRI 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 the imposition of technical specifications, and the parameters of the technical specifications (length, surveillance, etc.). The Reliability Assurance Program is expected to strongly influence the technical specifications.

Since the passive ALWR design philosophy departs from current licensing practices, new regulatory and review guidance is necessary so that the staff can appropriately review the AP-600 and SBWR submittals expected in mid-1992. Significant decisions need to be made concerning the scope of staff review of the non-safety systems and 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 that these active systems are available when needed in their defense-in-depth roles. The staff is conducting a sample evaluation of the passive designs to identify the safety importance of the passive safety and active non-safety systems. Using the results of the studies and interaction with EPRI, the staff will develop a review scope for these active non-safety systems, and will establish functional performance requirements and acceptance criteria to ensure they have adequate capability and availability when called upon.

The staff is still evaluating this issue for the passive plant designs. The discussion provided in this section is to provide a status of the issue for the Commission, and is provided to identify a complete list of issues applicable to the passive designs. The staff's proposed resolution of this issue will be provided to the Commission later. Therefore, the staff does not request any Commission action on this issue at this time.

B. Definition of Passive Failure

Appendix A to 10 CFR 50 states that any applicant must design against single failure of passive components in fluid systems important to safety, where a single failure is defined as an occurrence which results in the loss of capability of a component to perform its intended safety functions. Fluid and electric systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive component (assuming active components function properly) results in a loss of the capability of the system to perform its safety functions. However, the introduction to Appendix A to 10 CFR Part 50 notes that the conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development.

SECY-77-439 describes how the staff was using the single failure criteria in its reactor safety review process. As discussed in that paper, an active failure in a fluid system means (1) the failure of a component which relies on mechanical movement for its operation to complete its intended function on demand, or (2) an unintended movement of the component. Examples include the failure of a motor- or air-operated valve to move or to assume its correct position on demand; spurious opening or closing of a motor- or air-operated valve; or the failure of a pump to start or to stop on demand. In some instances, such failures can be induced by operator error.

A passive failure in a fluid system means a breach in the fluid pressure boundary or a mechanical failure which adversely affects a flow path. Examples include the failure of a simple check valve to move to its correct position when required; the leakage of fluid from failed components, such as pipes and valves - particularly through a failed seal at a valve or pump; or line blockage. Motor-operated valves which have the source of power locked out are allowed to be treated as passive components.

In past licensing reviews, the staff has been inconsistent in its treatment of passive failures in fluid systems in that in certain cases, it imposed a passive failure in addition to the initiating event while in others it did not. The staff has determined that in most instances, the probability of most types of passive failures in fluid systems is sufficiently small that they need not be assumed in addition to the initiating failure in application of the single failure criterion to assure the safety of a nuclear power plant.

In particular, staff practice has normally been to treat check valves, except for containment isolation systems, as passive devices rather than active devices during transients or design basis accidents. However, the staff is considering redefining failure of check valves to that of an active failure because safety-related check valves in the passive designs will operate under different conditions (low flow and pressure without pump pressure to open valves) than current generation reactors and evolutionary designs, they have increased safety significance to the operation of the passive safety systems, and operating experience has shown that they have a lower reliability than originally anticipated. This would cause these valves to be evaluated in a more stringent manner than that of previous licensing reviews.

The staff is still evaluating this issue for the passive plant designs. It is identifying this issue in this paper to comply with the Commission's directives to present policy matters for Commission consideration at the earliest opportunity. The staff's proposed resolution of this issue will be provided to the Commission later. Therefore, the staff does not request any Commission action on this issue at this time.

C. SBWR Stability

In BWRs, thermal-hydraulic instabilities can cause oscillations that can result in violation of the minimum critical power ratio (MCPR) safety limits. The staff has concluded that GE's analytical codes have been sufficiently validated to demonstrate the stability of the ABWR design. However, the codes that GE is using have not yet been adequately validated for the passive BWR design.

As discussed in SECY-91-273, the staff has determined that an early NRC assessment is needed of the vendor's analytical and experimental basis for demonstrating nuclear/thermal-hydraulic stability, and to identify any tests or analyses that may be needed to support staff technical evaluations of the issue. The NRC staff and its consultant, Oak Ridge National Laboratories (ORNL), have performed a review of the thermal-hydraulic stability characteristics of the SBWR based on preliminary design information provided by GE. This assessment included calculations with the LAPUR computer code developed by NRC/ORNL which showed that, while the system appears to be very stable under normal operating conditions, there are abnormal conditions of operation that might be reached under credible transient sequences that can result in the onset of density-wave power and flow oscillations. In addition, a low flow and low power instability due to a geysering effect between parallel channels has been identified as a concern during normal operating transients such as start-up and shutdown.

On December 6, 1991, the staff met with EPRI and GE to discuss the EPRI/GE response to the staff's conclusions that more extensive SBWR stability studies are needed and that codes which have been validated against thermal-hydraulic tests representative of the SBWR design, including the large open chimney, would be needed to perform these studies. EPRI and GE informed the staff that the chimney design has been changed and that existing experiments are repre-

sentative of the divided chimney now employed. GE indicated it will validate its codes for density-wave instability studies against these experiments and will provide the results of this work for NRC review. The geysering instability is being studied using small-scale experiments performed by a Japanese partner to GE. The Japanese SAFAR code will be validated against these experiments and used for analytical prediction of stable operating boundaries. GE plans to recommend start-up/shutdown procedures similar to those used in the Dutch Dodewaard reactor to avoid geysering instability. EPRI and GE believe that the SBWR is not vulnerable to a loop-type instability also reported by the Japanese; rather, it was characteristic of the experimental apparatus used.

GE has modified the SBWR conceptual design and has identified existing experimental data which they believe to be appropriate validation of codes to be used for stability studies. EPRI and GE have indicated agreement with the staff that such studies will be needed to confirm the stability behavior of SBWR during various transient scenarios (including ATWS). However, GE has not provided sufficient information to permit NRC evaluation of the applicability and sufficiency of the foreign experiments they have identified for use during code validation. The vendor has agreed to make this information available to the NRC as soon as it obtains permission from the foreign sources, possibly in early 1992. Until these experiments can be reviewed by NRC, the potential need for additional experiments to support stability evaluations for design certification remains open. The staff is identifying this issue in this paper to comply with the Commission's directives to present policy matters for Commission consideration at the earliest opportunity. The staff's proposed resolution of this issue will be provided to the Commission later. Therefore, the staff does not request any Commission action on this issue at this time.

D. Safe Shutdown Requirements

General Design Criterion (GDC) 34 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) are not exceeded. Regulatory Guide (RG) 1.139 and Branch Technical Position (BTP) 5-1 implement this requirement and set forth conditions to cold shutdown (200°F for a PWR and 212°F for a BWR) using only safety grade systems within 36 hours. The RG presents the basis for this requirement, as follows:

> ...even though it may generally be considered safe to maintain a reactor in a hot standby condition for a long time, experience shows that there have been events that required eventual cooldown and long-term cooling until the reactor coolant system was cold enough to perform inspection and repairs. It is therefore obvious that the ability to transfer heat from the reactor to the environment after a shutdown is an important safety function for both PWRs and BWRs. Consequently, it is essential that a power plant have the capability to go from hot-standby to cold-shutdown conditions...under any accident conditions.

Passive ALWR designs use passive heat removal systems for decay heat removal. They are restricted by the inherent ability of the passive heat removal processes and cannot reduce the temperature of the reactor coolant system below the boiling point of water for the heat to be transferred to the in-containment refueling water storage tank (IRWST) of the AP600 or the isolation condenser of the SBWR. Even though active shutdown cooling systems are available to bring the reactor down to cold shutdown or refueling conditions, these active RHR systems are not safety-grade, and do not comply with the guidance of RG 1.139 or BTP 5-1.

EPRI states that it is not necessary for the passive safety systems to be capable of achieving cold shutdown because they believe that the passive decay heat removal (DHR) systems have an inherent high long-term reliability. The EPRI Requirements Document for passive plant designs states that the passive ALWR designs will employ a redundant safety system, for both the hot standby and long-term cooling modes. In addition, it defines safe shutdown as 400°F. EPRI has indicated that it meets GDC 34 requirements because redundant passive decay heat removal systems can operate at full RCS pressure and place the reactor in the long-term cooling mode immediately after shutdown. Additionally, EPRI requires that operation of the plant in the long-term cooling mode be automatic, eliminating operator actions to cool down the plant. Operation of the passive DHR system also does not require any ac power or pumps to operate. EPRI further states that the non-safety systems that will take the plant to cold shutdown conditions "...are highly reliable in their own right...and a failure in these systems would not prevent the plant from achieving cold shutdown."

The staff is currently evaluating the EPRI position with respect to this issue to assess the acceptability of their proposed alternative approach for meeting GDC 34. The long-term DHR capability of the proposed passive systems offers potential advantages over current active systems. However, there are several issues that the staff must resolve before reaching a final position on this matter, including reliability criteria for the non-safety systems which have the capability to bring the plant to cold shutdown and the acceptability of 400°F as a safe long-term state. The staff is identifying this issue in this paper to comply with the Commission's directives to present policy matters for Commission consideration at the earliest opportunity. The staff's proposed resolution of this issue will be provided to the Commission later. Therefore, the staff does not request any Commission action on this issue at this time.

E. Control Room Habitability

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 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 Section 6.4 of the SRP, the staff defines this dose criterion in terms of specific whole body and organ doses (5 rem to whole body, and 30 rem each to thyroid and skin). Recently, the NRC has embraced the principal recommendations of Publication No. 26 of the International

Commission on Radiological Protection in the promulgation of a major revision of 10 CFR Part 20. The adoption of these recommendations, which includes use of the effective dose equivalent, did not result in a conforming change to the dose criteria for control room operators that is described in GDC 19. In current plants, safety-grade filtered control room HVAC systems with charcoal absorbers are used to ensure that radiation doses to operators could be maintained within the GDC 19 criteria in the event of an accident.

EPRI has proposed the exposure limit for control room operators to be 5 rem whole body, 75 rem beta skin dose, and 300 rem thyroid dose. For thyroid and beta skin doses, EPRI states that individual breathing apparatus and protective clothing will be provided if required to meet the regulatory limits. The staff has determined that EPRI's requirements for thyroid and beta skin dose (with no breathing apparatus and protective clothing) have not been adequately justified at this time. The permanent credit for the use of breathing apparatus in design basis accidents has never been allowed. The staff concludes that taking this credit is not acceptable because the long term use of breathing apparatus will degrade control room operator performance during and following an accident.

In addition, EPRI states that 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. 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 concludes that the analyses of control room habitability should consider the duration of the accident, and not just the EPRI-proposed 72-hour design basis period.

The EPRI Requirements Document states that the control room ventilation system for passive plants will be designed with appropriate redundancy and filtration (including charcoal), and will be powered both from on-site and off-site ac power sources. When ac power is available, control room habitability is provided by the non-safety grade heating, ventilation, and air conditioning (HVAC) system. However, EPRI states that ALWR designs should not rely upon the HVAC system to meet the current dose criteria described in Section 6.4 of the SRP. Instead, EPRI has proposed the less conservative dose criteria and the 72-hour design basis period.

The staff disagrees with EPRI on the above proposal. In order to resolve this matter, the staff may require that EPRI and the vendors provide a high level of assurance that the control room ventilation system will be available when needed. Although the 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 based on 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 as discussed in Section III.A of this enclosure. It should be noted that unlike the case of core-cooling, there is no passive safety-grade system to fall back on for control room habitability.

During a January 30, 1992 meeting with the staff, EPRI proposed an alternative approach to resolving this issue that conceptually uses a safety-grade pressurization system able to be recharged remotely after 72 hours to provide adequate ventilation for the control room operators during transients. The staff will evaluate EPRI's alternative proposal upon receipt.

As discussed in Section I.A of this enclosure, the staff is currently reviewing the new severe accident source term proposed by EPRI in conjunction with the staff's technical update effort on the TID-14844 source term. The estimated potential radiological consequences to the passive plant control room operators during a severe accident will depend on the outcome of the forthcoming resolution of severe accident source term, namely chemical forms of fission products, release fractions, and release timing. In addition, the control room habitability assessment is further dependent upon the fission product removal processes inside, as well as outside, of the primary containment before it reaches the control room air intake and/or the control building that houses the control room. Therefore, the staff is unable to complete its control room habitability assessment until the source term and its behavior mechanism are satisfactorily resolved.

However, the acceptable dose criteria for control room operators are not affected by the outcome of the source term review. Regardless of the outcome of the source term review, the staff concludes that the thyroid dose limit of 300 rem and the skin dose of 75 rem (with no protective clothing) specified in the Requirements Document without further technical justifications, does not meet the regulatory requirements and therefore, is not acceptable.

Therefore, the staff requests that the Commission approve the staff's positions that

1. the analyses of control room habitability be based on the dose criterion defined in GDC 19 of Appendix A to 10 CFR Part 50 and Section 6.4 of the SRP (5 rem to whole body, and 30 rem each to thyroid and skin), and

2. the analyses of control room habitability should be based on the duration of the accident in accordance with GDC 19 of Appendix A to 10 CFR Part 50.

In addition, the staff notes that the issue of control room habitability is closely tied with those concerning the revised source term and the regulatory treatment of non-safety systems. Should related policy questions be raised as a result of the staff's review of these matters, the staff will inform the Commission at the earliest opportunity.

F. Radionuclide Attenuation

EPRI and the passive ALWR designers rely on assumptions involving fission product removal processes inside containment by natural removal effects and holdup by the secondary building and piping systems. A containment spray system is not included in the EPRI Requirements Document for passive plant designs. The staff is concerned that use of the auxiliary building for holdup may require additional restrictions be placed on the auxiliary building during normal operation that the licensee may have difficulty complying with.

This issue also affects the control room habitability issue discussed in Section III.E of this report, as the industry indicates that fission products will be removed before they reach the control room air intake and/or the control building that houses the control room.

The staff is still evaluating this issue as well as the need for a containment spray system for the passive plant designs. The staff is also evaluating whether credit for the fission product attenuation in the main steamlines and condenser is appropriate for the passive BWR design because the main steamlines downstream of the main steam isolation valves and associated condenser are not designed to withstand the safe shutdown earthquake (SSE) as defined in Section III.c of 10 CFR Part 100. The staff concludes that plateout of radioactive iodine on the main steam pipe and condenser surfaces following a severe accident can realistically provide significant dose mitigation. Several technical references indicate that particulate and elemental iodines would be expected to deposit on surfaces with rates of deposition varying with temperature, pressure, gas composition, surface material, and particulate size.

These issues are identified in this paper to comply with the Commission's directives to present policy matters for Commission consideration at the earliest opportunity. The staff's proposed resolution of this issue will be provided to the Commission later. Therefore, the staff does not request any Commission action on this issue at this time.

G. Simplification of Off-Site Emergency Planning

EPRI has proposed to significantly simplify off-site emergency planning for passive designs because of EPRI's estimated low probability of core damage and, in the event of a core damage accident, the assurance of containment integrity and low off-site dose. EPRI's proposal would eliminate requirements for early notification of the public, detailed evacuation planning, and provisions for exercising the off-site plan. The on-site emergency plan and limited off-site actions would be retained. EPRI has identified this matter as a plant optimization subject.

During a January 30, 1992 meeting with the staff, EPRI proposed to work with the staff to define a process for addressing simplification of emergency planning, including developing technical criteria and methods that, if met,

would justify such action, and defining the process for implementation of this approach. The results of this effort would be used as input to a generic rulemaking proposal to be initiated by NUMARC.

The staff concludes that certain modifications from the emergency planning requirements of 10 CFR Part 50 and from the siting criteria in 10 CFR Part 100 may be appropriate for the passive designs based on their unique characteristics. However, an agency determination on these issues will require evaluation of detailed design information. The staff concludes that the unique characteristics of these designs should be taken into account in determining the extent of emergency planning requirements in the plume exposure pathway emergency planning zone. A plant's ability to prevent the significant release of radioactive material or to provide very long delay times prior to a release for all but the most unlikely events should be reflected in any decision on emergency planning requirements for the passive design. However, the staff will require a high degree of assurance that all potential containment bypass accident sequences have a very low likelihood before relaxing emergency planning requirements. The lack of information concerning source term and risk precludes further staff evaluation of the emergency preparedness requirements for the passive reactor designs at this time. Moreover, the issue is complicated by the fact that the promulgation of emergency planning requirements following the TMI-2 accident was not premised on any specific assumptions about severe accident probability. Hence it may be, as a policy matter, that even very low calculated probability values should not be a sufficient basis for changes to emergency planning requirements.

The staff will evaluate this issue for the passive plant designs when sufficient supporting information is available. It is identifying this issue in this paper to comply with the Commission's directives to present policy matters for Commission consideration at the earliest opportunity. The staff's proposed resolution of this issue will be provided to the Commission later. Therefore, the staff does not request any Commission action on this issue at this time.

2/11/92

COMMISSION PAPERS APPLICABLE TO ALWR ISSUES

Category	Issue No.	Issue Title	Commission Papers
I. SECY-90-016 Issues	A.	Use of Physically-Based Source Term	86-228 88-203 89-013 89-153 89-228 89-341 90-016 90-307 90-329 90-341 90-353
	B.	ATWS	89-153 89-228 90-016 90-353
	C.	Mid-Loop Operation	89-228 90-016 90-353
	D.	Station Blackout	89-013 89-153 89-228 90-016 90-329 90-353
	E.	Fire Protection	89-013 89-228 90-016 90-353
	F.	Intersystem LOCA	89-153 89-228 90-016 90-353

COMMISSION PAPERS APPLICABLE TO ALWR ISSUES (CONTINUED)

Category	Issue No.	Issue Title	Commission Papers
I. SECY-90-016 Issues (cont.)	G.	Hydrogen Control	89-013 89-153 89-228 90-016 90-329 90-353
	H.	Core-Concrete Interaction-Ability to Cool Core Debris	89-153 89-228 90-016 90-353
	I.	High Pressure Core Melt Ejection	89-228 90-016 90-353
	J.	Containment Performance	89-228 90-016 90-353 91-273
	K.	Dedicated Containment Vent Penetration	89-153 89-228 90-016 90-329 90-353
	L.	Equipment Survivability	89-228 90-016 90-353
	M.	Elimination of OBE	89-013 90-016 90-329 90-353 91-135
	N.	In-Service Testing of Pumps and Valves	89-228 90-016 90-353 91-273

COMMISSION PAPERS APPLICABLE TO ALWR ISSUES (CONTINUED)

Category	Issue No.	Issue Title	Commission Papers
II. Other Evolutionary and Passive Design Issues	A.	Industry Codes and Standards	91-273
	B.	Electrical Distribution	91-078
	C.	Seismic Hazard Curves and Design Parameters	91-135
	D.	Leak-Before-Break	89-013
	E.	Classification of Main Steamline of BWRs	none
	F.	Tornado Design Basis	none
	G.	Containment Bypass	none
	H.	Containment Leak Rate Testing	89-013 89-228 91-273
	I.	Post-Accident Sampling System	none
	J.	Level of Detail	90-241 90-377
	K.	Prototyping	91-074 91-273
	L.	ITAAC	91-178 91-210
	M.	Reliability Assurance Program	89-013
	N.	Site-Specific PRAs	89-013
	O.	SAMDAs	91-229
	P.	Generic Rulemaking Related to Design Certification	91-262

COMMISSION PAPERS APPLICABLE TO ALWR ISSUES (CONTINUED)

Category	Issue No.	Issue Title	Commission Papers
III. Passive Design Issues Only	A.	Regulatory Treatment of Non-Safety Systems	89-013 90-406
	B.	Definition of Passive Failure	77-439
	C.	SBWR stability	89-153 91-273
	D.	Safe Shutdown Requirements	none
	E.	Control Room Habitability	none
	F.	Radionuclide Attenuation	none
	G.	Simplification of Off-Site Emergency Planning	88-203

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SECY-89-153, "Severe Accident Design Features of the Advanced Boiling Water Reactor (ABWR)," May 10, 1989.

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SECY-90-341, "Staff Study on Source Term Update and Decoupling Siting from Design," October 4, 1990.

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SECY-90-406, "Quarterly Report on Emerging Technical Concerns," December 17, 1990.

SECY-91-074, "Prototype Decisions for Advanced Reactor Designs," March 19, 1991.

SECY-91-078, "Chapter 11 of the Electric Power Research Institute's (EPRI's) Requirements Document and Additional Evolutionary Light Water Reactor (LWR) Certification Issues," March 25, 1991.

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SECY-91-229, "Severe Accident Mitigation Design Alternatives for Certified Standard Designs," July 31, 1991.

SECY-91-262, "Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light Water Reactor (LWR) Designs," August 16, 1991.

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SECY-91-348, "Issuance of Final Revision to Appendix J to 10 CFR 50, and Reslated Final Regulatory Guide 1.XXX (MS 021-5)," October 25, 1991.

ACRS LETTER RELATED TO
DRAFT COMMISSION PAPER OF FEBRUARY 27, 1992
MAY 13, 1992

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

May 13, 1992

Mr. James M. Taylor
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Taylor:

SUBJECT: ISSUES PERTAINING TO EVOLUTIONARY AND PASSIVE LIGHT WATER REACTORS AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

During the 383rd, 384th, and 385th meetings of the Advisory Committee on Reactor Safeguards, March 5-7, April 2-4, and May 6-9, 1992, we discussed with representatives of the NRC staff the staff's positions, recommendations, and resolution schedules concerning the certification issues for evolutionary and passive light water reactors contained in the draft SECY paper dated February 7, 1992. We also had the benefit of the documents referenced. The staff requested ACRS comments on the draft SECY paper. Our comments and recommendations on some of the staff's positions are given below.

I. SECY-90-016 Issues

Item M. Elimination of Operating Basis Earthquake

Appendix A to 10 CFR Part 100 currently establishes the Operating Basis Earthquake (OBE) at a level one-half of the Safe Shutdown Earthquake (SSE). With this specification, the OBE exerts undue influence over the seismic design and requires a full spectrum analysis in addition to that of the SSE. The staff's proposal is to effectively decouple the OBE from design. We agree with the staff's recommendation.

II. Other Evolutionary and Passive Design Issues

Item A. Industry Codes and Standards

We agree with the staff's recommendation to use the newest codes and standards that have been endorsed by the NRC in its reviews of both the evolutionary and passive plant design applications, and its

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recommendation that unapproved revisions to codes and standards be reviewed on a case-by-case basis.

Item D. Leak Before Break

We agree with the staff's recommendation to extend the application of the leak-before-break approach for both evolutionary and passive advanced light water reactors.

Item E. Classification of Main Steamlines of Boiling Water Reactors (BWRs)

We agree with the staff's recommendation for resolution of the main steamline classification for both evolutionary and passive BWRs.

Item F. Tornado Design Basis

Based on a study (NUREG/CR-4661) that compiled a considerable quantity of tornado data, the staff recommends that the maximum tornado wind speed of 300 mph (compared with the present 360 mph) be used for the design-basis tornado. We agree that the best available data should be used, but caution that design-basis specifications have sometimes been established conservatively to provide margins to deal with events not specifically addressed in the design basis. We recommend that the staff's position be approved with a qualification that the staff require assurance that other potential loads that may have been previously subsumed within the tornado design basis be taken into account if necessary.

Item H. Containment Leakage Rate Testing

The staff recommends that the maximum interval between Type C leakage rate tests for both evolutionary and passive designs be increased to a 30-month interval from the 24-month interval now required in 10 CFR Part 50, Appendix J. No significant safety penalty caused by this change has been identified. We agree with the proposed staff position.

Item I. Post-Accident Sampling System (PASS)

The staff is requesting approval of changes in requirements for the PASS currently found in 10 CFR 50.35(f)(2)(viii). These requirements, and the

guidance contained in Regulatory Guide 1.79 and in NUREG-0737, resulted from consideration of the TMI-2 accident.

We agree with the staff's proposal but have the following comments:

1. The requirements as contained in the above referenced regulation refer to "the reactor coolant system and containment that may contain TID-14844 source term radioactive materials" and to measurement of these and other materials. In light of source terms now considered in severe accident analysis, it is advisable to revise this obsolete description.

2. The proposal for "Elimination of the Hydrogen Analysis of Containment Atmosphere Samples" is appropriate, given that safety grade hydrogen monitoring instrumentation will be installed.

3. The Electric Power Research Institute (EPRI) proposed elimination of an existing requirement for the capability to sample the reactor coolant at operating pressure in order to measure the dissolved gas and chloride in the coolant. EPRI claims that maintaining the systems on existing plants produces significant exposure of operating personnel, and that given a severe accident, no useful information, not otherwise available, is provided by this capability. The staff proposes to retain the requirement, but to change the time after accident onset at which the capability must be available from 8 to 24 hours. During our discussion with the staff, we were unable to elicit any reason for this requirement other than that it was established following the TMI-2 accident. We cannot endorse continuation of the requirement for high pressure sampling on the basis of information available to us.

4. The staff proposes approval of a position that "would require the capability to take samples for boron and for activity measurements 8 hours and 24 hours, respectively, after the end of power operation." The intent appears appropriate, however, we suggest that it might be better to specify a time at which the information from measurements becomes avail-

able to the operator rather than the time at which samples can be taken. Further, we assume that what is required is boron concentration rather than the presence or absence of boron. Finally, we suggest that the phrase "after the end of power operation" be made more specific.

Item N. Site-Specific Probabilistic Risk Assessment

If, as concluded by the staff, enveloping analyses are practical for both seismic events and tornadoes, it is appropriate that these be part of the submittal at the time of certification. However, enveloping analyses are not as practical for other external events such as river flooding, storm surge, tsunamis, hurricanes, and volcanism. Therefore, the staff recommends that these other types of site-specific PRA information be submitted at the combined operating license (COL) stage. We agree with this recommendation but would like to hear more about how the staff proposes to deal with any unacceptable findings at the COL stage.

Sincerely,

David A. Ward
Chairman

References:

1. Draft SECY paper dated February 7, 1992, for the Commissioners, from James M. Taylor, NRC Executive Director for Operations, Subject: Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements (Draft Predecisional)
2. SECY-90-016 dated January 12, 1990 for the Commissioners from James M. Taylor, NRC Executive Director for Operations, Subject: Evolutionary Light Water Reactor (LWR) Certification Issues and their Relationship to Current Regulatory Requirements
3. U.S. Nuclear Regulatory Commission, NUREG/CR-4661, Subject: Tornado Climatology of the Contiguous United States, dated May 1986

STAFF RESPONSE TO ACRS LETTER
OF MAY 15, 1992
JUNE 12, 1992

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

June 12, 1992

Mr. David A. Ward, Chairman
Advisory Committee on Reactor Safeguards
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Ward:

SUBJECT: ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS) COMMENTS REGARDING THE DRAFT COMMISSION PAPER, "ISSUES PERTAINING TO EVOLUTIONARY AND PASSIVE LIGHT WATER REACTORS AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS"

In your letter of May 13, 1992, you provided the U. S. Nuclear Regulatory Commission (NRC) staff with ACRS comments regarding the NRC staff's positions on several of the issues discussed in the draft Commission paper, "Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements."

The NRC staff notes that the ACRS agrees with the staff's proposed positions on several of the issues discussed in the draft paper. We also note the ACRS' comments and recommendations on the tornado design basis (Item II.F), post-accident sampling system (PASS) (Item II.I), and site-specific probabilistic risk assessment (PRA) (Item II.N) issues. We will consider your comments as we revise and finalize the subject Commission paper.

In Item II.F of the draft Commission paper, the staff stated that it will ensure that other loads that may have been enveloped within the tornado design basis will be identified and, if necessary, considered in the design basis. The staff will ensure that, if an actual site hazard exceeds the design envelope in a certain area, the applicant will perform an analysis to verify that the design remains acceptable for the site.

The staff has reviewed its basis for requiring that the PASS have high pressure sampling capability in order to measure the concentrations of dissolved gas and chloride in the reactor coolant. The staff believes that the need for high pressure sampling stems from the possibility of partially mitigated severe accidents which do not involve early reactor depressurization. This is fundamentally a PWR concern since, for a PWR, there appears to be a substantially greater probability of a severe accident being arrested while the reactor coolant system remained at pressure. In fact, the TMI-2 accident is just such an example. In addition, there is a significantly greater concern for maintaining reliable natural circulation and decay heat removal in the presence of the gases which would evolve as a PWR was depressurized. The evolution of a large amount of non-condensable gas in a passive PWR would appear to be the most sensitive scenario since the passive plant decay heat removal systems are so dependent on natural circulation. It should also be noted that the passive safety systems on a passive PWR would not take the plant to a cold shutdown condition following a partially mitigated severe

accident. The active non-safety systems would need to be used by the operators to perform the final cooldown and depressurization. When such actions are called for, the staff believes that the operators need to have a full appreciation of the consequences of depressurizing the plant and possibly introducing non-condensable gases which could complicate or interfere with the successful termination of the event. The operator can then make informed decisions relating to issues such as: reactor coolant pump start-up and operation; instrumentation reliability; and the use of active or passive decay heat removal systems.

For the longer term, determining the oxygen and chloride concentrations would help ensure that plant personnel take appropriate actions to minimize the likelihood of accelerated primary system corrosion following the accident. This is a secondary consideration since long term samples could likely be taken at low pressure. However, once the need to take the high pressure sample has been established, because of the non-condensable gas concern, it does appear appropriate to test such a sample for oxygen and chlorides in order to get an early indication of potential corrosion problems. Therefore, the staff has concluded that PASS high pressure sampling capability is necessary. The staff has reviewed the proposed requirement for boron sampling and will clarify the requirement to specify that the analyses include determination of the boron concentration. In addition, the staff will clarify the term "end of power operations" to help ensure that the sampling time requirements are clearly understood.

The staff is developing a position on the treatment of external events in site-specific PRAs for advanced light water reactor designs. The staff will incorporate this position into a draft Commission paper which is scheduled to be forwarded to the ACRS by July 1992. The staff plans to develop a separate Commission paper discussing the form and content of the combined license (COL). This Commission paper should help address any questions the ACRS may have on resolving findings made during the COL review process.

Sincerely,

/s/

James M. Taylor
Executive Director
for Operations

cc: The Chairman
Commissioner Rogers
Commissioner Curtiss
Commissioner Remick
Commissioner de Planque
SECY

DISTRIBUTION:
See next page

ACRS LETTER RELATED TO
DRAFT COMMISSION PAPER OF FEBRUARY 27, 1992
AUGUST 17, 1992

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

August 17, 1992

Mr. James M. Taylor
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Taylor:

SUBJECT: ISSUES PERTAINING TO EVOLUTIONARY AND PASSIVE LIGHT WATER REACTORS AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

During the 386th, 387th, and 388th meetings of the Advisory Committee on Reactor Safeguards, June 4-5, July 9-11, and August 6-8, 1992, we discussed with representatives of the NRC staff the staff's positions, recommendations, and resolution schedules concerning the certification issues for evolutionary and passive light water reactors contained in the draft SECY paper dated February 7, 1992. This supplements our letter of May 13, 1992, and provides our comments and recommendations on some of the staff's positions for the passive light water reactors. The section titles and letter designations correspond to those in the draft SECY paper.

I. SECY-90-016 Issues (For Passive Plants)

E. Fire Protection

The NRC staff is seeking Commission approval to use the enhanced fire protection criteria previously approved for evolutionary Advanced Light Water Reactor (ALWR) plants by the Commission's Staff Requirements Memorandum (SRM) of June 26, 1990. This SRM approved the staff's position on fire protection as presented in SECY-90-016 and supplemented by the staff's April 27, 1990 response to our report on the SECY. We recommended separate Heating, Ventilating, and Air Conditioning (HVAC) systems for each division as an important step toward ensuring adequate environmental separation of safety systems. The staff agreed that consideration of smoke, heat, and fire suppressant migration may result in separate HVAC systems, but other options may be available to the designer. Our report to the Commission of April 13, 1992, on the Draft Safety Evaluation Report for the ABWR identified the adequacy of physical separation as a continuing issue for the

ABWR, due in part to the use of a shared HVAC system for multiple trains of redundant safety systems during normal plant operation.

Our concern with shared HVAC systems is related to the need for adequate isolation of such systems during certain disruptive events (e.g., fires, floods, or pipe breaks). If the isolation is not adequate, the HVAC arrangement may become a pathway whereby effluents from the event are conducted to locations where required safe shutdown equipment is located. This is not a concern if either (1) the HVAC isolation provisions are able to withstand the event consequences (e.g., pipe whip, jet impingement, static and dynamic pressure, and elevated temperature) during and after closure with consideration of single active component failures and acceptable leakage, or (2) the safe shutdown equipment is qualified for the environmental exposure resulting from a release of the adverse environment at any credible location along the HVAC pathway such as duct openings or blowout locations.

Except for the concern with shared HVAC, we support the staff recommendation that the passive plants should be reviewed against the enhanced fire protection criteria approved in the Commission's SRM.

F. Intersystem Loss-of-Coolant-Accident

The staff's position is that designing these low-pressure fluid systems that interface the reactor coolant system (RCS) to withstand full RCS pressure (to the extent practicable) is an acceptable means for resolving this issue. For those systems that have not been designed to withstand full RCS pressure, the staff indicates that other measures will be required. We recommend approval of the proposed staff resolution, provided consideration is given to all elements of the low pressure piping system (e.g., instrument lines, pump seals, heat exchanger tubes, and valve bonnets).

G. Hydrogen Control

The staff recommends that the evolutionary LWR designs provide a system for hydrogen control that can safely accommodate hydrogen generated by the reaction of steam with 100 percent of the fuel cladding surrounding the active fuel. (Note: This is not 100 percent of the reactive metal in the core.) We support the staff's recommendation.

The staff also recommends that the system be capable of precluding uniform containment concentrations of hydrogen greater than 10 percent. We are aware of analytical work in

support of the resolution of Generic Issue 106, "Piping and the Use of Highly Combustible Gases in Vital Areas," that suggests the possibility of transition to detonation at average concentrations as low as 12 percent. We recommend that the staff do a similar analysis of the impact of hydrogen combustion, and possible detonation including stratification, before establishing a limit for the average hydrogen concentration. This is of particular importance to steel-shell containments.

I. High Pressure Core Melt Ejection

To cope with the possible effects of direct containment heating (DCH), the staff concludes, ". . . that ALWR design should include a depressurization system and cavity design features to contain ejected core debris."

DCH is an extremely improbable event, and we see no need to require two modes of coping with the possibility. Either depressurization or cavity design provisions alone should be adequate. Because of possible safety benefits for other events, reliable depressurization is the preferred approach.

J. Containment Performance

The staff has not yet developed an adequate technical position relating to requirements for containment performance in passive LWRs. We agree that the proposed value of 0.1 for a conditional containment-failure probability (CCFP) is reasonable but, as we stated in our letter of April 26, 1990, regarding "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," this value is defined only within the context of a family of initiating events. It should be used by the staff in the development of its requirements and not merely passed on to applicants.

The deterministic criterion proposed by the staff is not a simple alternative to the CCFP. It could be used more logically as a complement. Using ASME Code Service Level C stress limits is not unreasonable given a known loading for which the containment is to be designed. However, determination of the appropriate loading is the hard part of the problem and the suggested deterministic criterion is essentially meaningless without it. The staff states that "applicants using the deterministic approach will be required to define the challenges considered in this evaluation." The staff takes no position on what those challenges should be or how they are to be quantified. Apparently the intent is to default to a "design specific review." This approach leaves

the applicant without any real guidance from the Commission on this important topic.

We acknowledge that it is a very difficult task to establish containment performance criteria but is important. We suggested what we believe to be the best approach in our letter of May 17, 1991, "Proposed Criteria to Accommodate Severe Accidents in Containment Design."

K. Dedicated Containment Vent Penetration

The staff proposes that the decision on the need for a containment vent for passive designs should not be made at this time but should wait until specific plant designs are evaluated. We believe that the Commission should make a generic judgment about the acceptability of containment vents for LWRs. This should be a part of establishing general criteria for containment design as proposed in our letter of May 17, 1991.

L. Equipment Survivability

We agree with the staff's recommendation that features provided only for severe-accident mitigation for the passive plant designs not be subject to the environmental qualification requirements of 10 CFR 50.49, quality assurance requirement of 10 CFR 50, Appendix B, and redundancy/ diversity requirements of 10 CFR 50, Appendix A.

N. In-Service Testing of Pumps and Valves

We support the staff recommendation that the special pump and valve design, testing, and inspection provisions be imposed on all safety-related pumps and valves for the passive ALWRs.

III.E - Control Room Habitability

There were several significant differences between the staff and EPRI at the time the staff drafted this policy issue. EPRI has subsequently made a proposal to modify its Utility Requirements Document to include a requirement for a passive, safety grade, control room pressurization system that would use a bottled air supply to maintain operator doses within regulatory limits for the first 72 hours following an accident. (The regulations require that operator doses be so limited for the duration of the accident.) The pressurization system proposed by EPRI would be designed to be replenished by off-site portable supplies after 72 hours if needed. Accordingly, EPRI has recommended that the staff close this issue.

We discussed this matter with the staff and EPRI during our June 4-5, 1992 meeting. The staff told us that it is currently evaluating the EPRI proposal and is not prepared to close this issue. ACRS had several comments regarding design features of the passive control room pressurization system proposed by EPRI. We believe that the staff should take these comments into account in its evaluation. We may provide additional recommendations after the staff has completed its evaluation.

Sincerely,

David A. Ward
Chairman

References:

1. Draft SECY Paper dated February 7, 1992, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements
2. SECY-90-016 dated January 12, 1990, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements
3. Memorandum dated April 27, 1990, from James M. Taylor, Executive Director for Operations, NRC, for NRC Commission, Subject: Staff Response to ACRS Conclusions Regarding Evolutionary Light Water Reactor Certification Issues

ANNEX D

DRAFT COMMISSION PAPER
"DESIGN CERTIFICATION AND LICENSING POLICY ISSUES
PERTAINING TO PASSIVE AND EVOLUTIONARY ADVANCED LIGHT WATER REACTOR DESIGNS"
JULY 6, 1992

DRAFT

For: The Commissioners

From: James M. Taylor
Executive Director for Operations

Subject: DESIGN CERTIFICATION AND LICENSING POLICY ISSUES PERTAINING TO PASSIVE AND EVOLUTIONARY ADVANCED LIGHT WATER REACTOR DESIGNS

Purpose: To present the Commission with several additional issues that the staff has identified pertaining to passive and evolutionary light water reactors (LWRs) and to request that the Commission approve the positions described in this paper. The staff will review those issues for which it has not recommended a position, and will provide the Commission with a recommendation when the review is complete.

Summary: The staff has discussed seven additional technical and policy issues pertaining to either evolutionary LWRs, passive LWRs or both. In addition, the staff has provided the results of its review concerning the regulatory treatment of nonsafety-related systems in passive plant designs. The staff has underlined the positions for which it is requesting Commission's approval.

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DRAFT

Background: The staff continues to inform the Advisory Committee on Reactor Safeguards (ACRS) and the Commission of new policy issues early in the review process, so that the Commission can determine the approach for resolving them. The staff is providing an analysis of the areas in which it proposes to depart from current regulations or in which it is supplementing or revising interpretive guidance applied to currently licensed LWRs.

The staff has forwarded several policy papers[1] to the Commission proposing resolutions for policy matters and major technical issues concerning both the evolutionary and passive LWR designs. The draft Commission paper, "Issues Pertaining to Evolutionary and Passive Light Water Reactors and their Relationship to Current Regulatory Requirements," was released to the public 7 days after it was forwarded to the Commission to support dialogue between the NRC staff, the ACRS, the EPRI, the vendors, and other industry representatives.

Similarly, the staff proposes to release this draft Commission paper and Enclosures 1 through 3 to the public 3 days after forwarding it to the Commission.

The discussions contained in this draft Commission paper may be combined with the discussions contained in the draft Commission paper, "Issues Pertaining to Evolutionary and Passive Light Water Reactors and their Relationship to Current Regulatory Requirements," and forwarded to the Commission as a single Commission paper.

Discussion: The staff has identified significant issues regarding the safety of evolutionary and passive advanced light water reactor (ALWR) designs. The staff proposed resolutions for some of these issues in Commission papers SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," SECY-91-078, "Chapter 11 of the Electric Power Research Institute's (EPRI's) Requirements Document and Additional Evolutionary Light Water (LWR) Certification Issues," and the draft Commission paper which was forwarded to the Commission on February 20, 1992, "Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements." The Commission

[1] Enclosure 3 is a list of the papers that the staff has forwarded to the Commission regarding policy issues that have been identified for evolutionary and passive ALWRs. The staff will reference applicable documents throughout this paper.

addressed SECY-90-016 and SECY-91-078 in its SRM of June 26, 1990, and August 15, 1991, respectively.

The issues discussed in Enclosure 1 of this paper include

A. defense against common mode failures in digital instrumentation and control (I&C) systems
B. analysis of external events beyond the design basis
C. elimination of the operating basis earthquake from seismic design
D. multiple steam generator tube ruptures
E. probabilistic risk assessment (PRA) beyond design certification
F. role of a passive plant control room operator
G. control room annunciator reliability
H. regulatory treatment of nonsafety systems in passive plant designs

The issues concerning the role of the operator in a passive plant control room and the regulatory treatment of nonsafety systems apply only to passive plant designs. The six remaining issues apply to evolutionary and passive ALWR designs. Background information for some of these issues can be found in the previous Commission papers that are referenced in Enclosures 1 and 2.

Enclosure 1 provides a description of the eight policy issues including a discussion of the current regulatory requirement or interpretation, a discussion of the industry position and, where available, a discussion of the staff's proposed position. The staff has included a detailed discussion of the basis for the staff's position on each issue, as available. The staff has underlined the positions for which it is requesting the Commission's approval. Issues for which no resolution has been proposed will be addressed in the future.

Enclosure 2 is a list of issues pertaining to evolutionary, passive, and advanced reactor designs. Adjacent to each issue is a list of Commission papers in which the staff has addressed the issue. Enclosure 3 is a list of Commission papers which concern ALWR designs.

The staff developed the recommendations in this paper after (1) reviewing current operating reactor designs, evolutionary designs, and limited passive ALWR design information; (2) considering operating experience; (3) evaluating the results of the PRAs of LWRs and ALWRs; (4) considering the Commission's guidance on issues resolved for the evolutionary ALWRs; (5) completing the draft safety evaluation

reports for the EPRI Utility Requirements Document for passive and evolutionary ALWR designs; and (6) considering EPRI and vendor comments on these issues.

The staff considered EPRI comments regarding aspects of these additional policy issues discussed in letters from EPRI of April 9, May 5, and May 8, 1992. In its letter of April 9, 1992, EPRI provided a detailed discussion on defense against common mode failure in digital instrumentation and control systems. In its letter of May 18, 1992, Asea Brown Boveri/Combustion Engineering discussed diversity for the digital instrumentation system for its System 80+ design.

The staff concludes that these issues are fundamental to the Agency's decisions on the acceptability of the evolutionary and passive designs. As discussed in SECY-91-262, "Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light Water Reactor (LWR) Designs," the staff proposes to implement final positions on these matters as approved by the Commission through individual design certifications and generic rulemaking, as appropriate, for the evolutionary and passive designs.

The staff concludes that it would be beneficial to provide EPRI and the ALWR vendors with the staff's conclusions presented herein as expeditiously as possible in order to help resolve those matters concerning both the evolutionary and passive reactor designs, and to facilitate further discussion of the issues for the passive designs.

Therefore, the staff proposes to issue Enclosures 1 through 3 to the industry after 3 working days from the date this paper is forwarded to the Commission to support meetings between the NRC staff, EPRI, the vendors, and the ACRS. The staff will indicate that the proposed resolutions are before the Commission for consideration, and, therefore, may not be final positions.

The staff does not expect to identify additional policy matters for the evolutionary designs. However, as the staff proceeds with its review of the EPRI ALWR Requirements Document for passive plants and the passive LWR design applications, it may find other issues not discussed herein. The staff allowed time for identifying and addressing such issues in the schedule estimates for the passive reactor design reviews provided in SECY-91-161, "Schedules for the Advanced Reactor Reviews and Regulatory Guidance Revisions." The staff will notify the Commission of new policy matters when they are found.

Conclusions: The staff requests approval of the staff's proposed positions for those issues pertaining to evolutionary LWR designs in order to perform the final design approval and the design certification review of General Electric's Advanced Boiling Water Reactor and Asea Brown Boveri (ABB) Combustion Engineering's System 80+ LWR designs.

The staff also requests that the Commission approve the staff's proposed positions for those issues pertaining to the passive designs. This will enable the staff to proceed more efficiently with its schedule review of Westinghouse's Advanced Passive AP600 and General Electric's Simplified Boiling Water Reactor ALWR designs.

Coordination: The Office of General Counsel (OGC) has reviewed this paper and has no legal objection. OGC notes that Commission approval would be tentative, subject to further review in design certification rulemakings and that communications with ALWR vendors and EPRI regarding these Commission positions should state this fact. The staff is forwarding this paper to the ACRS for its review and comments.

Recommendations: That the Commission

(1) Approve the staff's positions underlined in Enclosure 1.

(2) Note that the staff will seek the Commission's approval of its positions for those issues that the staff is reviewing before taking a final position with the applicant or industry.

(3) Note that the staff will provide Enclosures 1 through 3 to the ALWR vendors and EPRI after 3 working days from the date this paper is forwarded to the Commission. The staff will indicate that the proposed resolutions are before the Commission for consideration, and, therefore, may not be final positions.

DRAFT

(4) Note that if the staff finds other policy issues, it will seek the Commission's approval of its positions in a timely manner.

James M. Taylor
Executive Director
for Operations

Enclosures:
1. Policy Issues Analysis and Recommendations
2. ALWR Issue Cross-reference Matrix
3. Commission Papers Applicable to ALWRs

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Enclosure 1

POLICY ISSUES ANALYSIS AND RECOMMENDATIONS

The following is a discussion of eight design certification and licensing issues associated with the review of evolutionary and advanced light water reactor (ALWR) designs. The staff has included a detailed discussion of the basis for each of its proposed positions that depart from current regulatory requirements. The staff has underlined the positions for which it is requesting the Commission's approval. Issues for which no resolution has been proposed will be addressed in the future.

A. Defense Against Common Mode Failures in Digital Instrumentation and Control Systems

Instrumentation and control (I&C) systems help to ensure that the plant operates safely and reliably by monitoring, controlling, and protecting critical plant equipment and processes. The digital I&C systems for ALWRs differ significantly from the analog systems used in operating nuclear power plants. The digital I&C system shares more data transmission functions and process equipment than an analog system would share.

Digital I&C systems share databases (software) and process equipment (hardware). Therefore, a hardware design error, a software design error, or a software programming error may result in a common mode failure of redundant equipment. The staff is concerned that the use of digital computer technology in I&C systems could result in safety-significant common mode failures. The staff developed these concerns more fully in SECY-91-292, "Digital Computer Systems for Advanced Light Water Reactors." Some of the major points contained in SECY-91-292 are summarized as follows:

- Common mode failures could defeat not only the redundancy achieved by the hardware architectural structure but also could result in the loss of more than one echelon of defense in depth provided by the monitoring, control, reactor protection, and engineered safety functions performed by the digital I&C systems.

- The two principal factors for defense against common mode failures are quality and diversity. Maintaining high quality will increase the reliability of both individual components and systems. Diversity in assigned functions, for both equipment and human activities, and diversity in hardware and software can reduce the probability that a common mode failure will propagate.

- The staff intends to require some level of diversity, such as a reliable analog backup.

Current regulations applicable to analog I&C systems also apply to digital I&C systems. In addition, the staff has developed limited guidance for digital I&C systems in Regulatory Guide 1.152, "Criteria for Programmable Digital Computer Software Systems in Safety Related Systems of Nuclear Power Plants."

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However, as discussed in SECY-91-292, there are currently no regulatory requirements which adequately address the potential safety concerns associated with digital I&C systems. Quality and diversity are important defenses against common mode failures. However, there are no consensus standards for certifying the design of digital I&C systems for application in nuclear power plant designs. In Enclosure 2 to SECY-91-292, the staff discussed regulatory requirements that it is considering to help ensure defense against common mode failures. The staff is considering new requirements in the following areas:

- assessment of diversity
- engineering activities
- design implementation
- safety classification of I&C systems

The staff has made significant progress in establishing regulatory guidance that could be used for assessing diversity. The staff, with the support of Lawrence Livermore National Laboratory (LLNL), has performed a study of the General Electric Company (GE) advanced boiling water reactor (ABWR) design to assess the adequacy of its defense in depth and diversity. This assessment was performed using the method described in NUREG-0493, "A Defense-in-Depth and Diversity Assessment of the RESAR-414 Integrated Protection System" for each transient and accident evaluated in Chapter 15 of the safety analysis report. The staff will use the results of this assessment to help determine if additional diversity is necessary to defend against postulated common mode software and hardware failures.

EPRI discussed requirements for engineering activities and design implementation for digital I&C systems in Chapter 10 of the EPRI Advanced Light Water Reactor Utility Requirements Documents (URDs) for both the evolutionary and the passive plants (Volumes II and III, respectively). The staff has discussed the issue of diversity in digital control systems with EPRI. The staff concludes that the EPRI requirements do not contain sufficient detail to help ensure adequate diversity. In SECY-91-292, the staff discussed regulatory requirements that could be developed for engineering activities and design implementation.

As discussed in SECY-91-292, the staff is continuing to develop safety classification criteria for I&C systems in ALWR designs. As discussed in SECY-91-292, the international technical community is considering this topic in the draft International Electrotechnical Commission (IEC) standard, "The Classification of Instrumentation and Control Systems Important to Safety for Nuclear Power Plants." EPRI proposed certain classification standards in its "ALWR Position Paper for Passive System Classification and Requirements," submitted by letter dated March 19, 1992. The staff will consider these positions before reaching a final position on safety classification criteria for I&C systems in ALWR designs. The safety classification of digital I&C systems relates to diversity through the defense-in-depth assessment of safety credit for systems that have previously been classified as nonsafety systems.

Recently, increased attention has been given to detailed assessments of the integrity of software for safety-critical functions. These assessments have covered a broad range of applications which include computer-based medical treatment facilities, computer-based fly-by-wire aircraft control systems, and protection systems for nuclear power plants. The staff found a consensus among computer science and software engineering experts that applications critical to safety should be backed-up by some system not based on software because the quantitative estimate for the reliability of I&C systems based on high integrity software cannot yet be determined. The type of this backup and the extent of the functions it should perform depend on the level of confidence in the computer systems.

As previously discussed, the staff has made significant progress in establishing potential regulatory guidance to ensure adequate diversity for digital I&C system applications. As a result of its review, the staff recommends that the Commission approve the following approach for assessing diversity and the following requirements for a backup system which is not based on software and which is to be used for systems-level actuation and displays:

1. The applicant shall assess the defense in depth and diversity of the proposed instrumentation and control system to demonstrate that vulnerabilities to common mode failures have been adequately addressed. The staff considers software design errors to be credible common mode failures that must be specifically included in the evaluation. An acceptable method of performing analyses is described in NUREG-0493. Other methods proposed by an applicant will be reviewed individually.

2. In performing the assessment of defense in depth and diversity, the vendor or applicant shall analyze each postulated common mode failure for each event that is evaluated in the accident analysis section of the safety analysis report (SAR). The vendor or applicant shall demonstrate adequate diversity within the design for each of these events.

3. If a postulated common mode failure could disable a safety function, then a diverse means, with a documented basis that the diverse means is unlikely to be subject to the same common mode failure, shall be required to perform either the same function or a different safety function that provides equivalent protection. The diverse or different safety function may be performed by a nonsafety system if the system is of sufficient quality to perform the necessary function under the associated event conditions. Diverse digital or non-digital systems are considered to be acceptable means. Manual actions from the control room are acceptable if time and information are available to the operators. The amount and type of diversity may vary among designs and will be evaluated individually.

4. A set of safety-grade displays and controls, independent of the computer system(s) and located in the main control room, shall be provided for system-level actuation and monitoring of critical safety functions and

parameters. The displays and controls shall be provided for those system-level actuations and critical safety functions and parameters which are required by control room operators to place the reactor plant in a hot shutdown condition. The displays and controls shall be conventionally hardwired in the system architecture to the lowest level practicable. Each set of equipment required will be evaluated individually.

The hardwired system-level controls and displays provide the plant operators with unambiguous information and control capabilities. These hardwired controls and displays are required to be in the main control room to enable the operators to expeditiously mitigate the effects of the postulated common mode failure of the digital I&C system. The control room would be the center of activities to safely cope with the event which could also involve the initiation and implementation of the plant emergency plan. The design of the plant should not require operators to leave the control room for such an event. For the longer term recovery operations, credit may be taken for actions from outside the main control room, when the emergency response organization is fully briefed and in place to take such actions.

B. Analyses of External Events Beyond the Design Basis

In the Commission policy statement, "Policy Statement on Severe Accidents Regarding Future Designs and Existing Plants," issued on August 8, 1985 (50 FR 32138), the Commission stated that applicants for future evolutionary reactor plant design reviews should complete a probabilistic risk assessment (PRA) and consider improving the means to avoid or mitigate severe accident vulnerabilities exposed by the PRA in order to help ensure the public health and safety. The Commission also stated that evolutionary plant vendors should use the PRA in considering a range of alternatives and a combination of alternatives that address unresolved and generic issues and to search for cost-effective means to reduce the risk from severe accidents. In the policy statement, the Commission stated that the staff should review evolutionary ALWR designs to determine the safety acceptability of the design, stressing deterministic engineering analysis and judgment, complimented by PRA. After issuing this policy statement, the Commission incorporated 10 CFR Part 52 into the regulations. Section 52.47 of 10 CFR Part 52 requires that an application for design certification contain a design-specific PRA.

In Generic Letter 88-20, "Individual Plant Examinations for Severe Accident Vulnerabilities - 10 CFR 50.54(f)," and its supplements, the staff states that construction permit holders and power reactor licensees should consider the safety implications of both internal and external events by performing separate individual plant examinations (IPEs) and individual plant examinations for external events (IPEEE). PRAs and IPEs that have evaluated both internal and external events generally estimate the risks from external events to be the same order of magnitude as internal events. Therefore, the staff concludes that the design-specific PRAs required in 10 CFR Section 52.47 should include an assessment of both internal and external events.

Lessons from past studies indicate that fire, internal floods, and seismic events can be important contributors to core damage. However, the estimates of core damage frequencies for the fire and seismic events continue to include considerable uncertainty. Therefore, the staff concludes that fire and seismic events can best be evaluated using the margins methods developed for existing plants, supported by insights from internal events PRAs, to find the vulnerabilities of these new designs that could be important to safety. Following this method, the designer would focus on the capacity of the design to sustain the effect of the external events, rather than relying on the highly uncertain estimates. The designer could use traditional probabilistic techniques to study internal floods.

The staff intends to determine the adequacy ("robustness") of ALWR designs by having the designer perform a modified margins analysis to determine the vulnerabilities to seismic events. The designer can best determine the seismic capability of the plant by merging PRA and margins approaches to take advantage of the strengths of each. This approach allows for a comprehensive and integrated treatment of the plant's response to an earthquake. Plant logic models covering the various systems that could be used to prevent core damage are constructed, typically, by modifying PRA models for internal events to include logic important in considering seismic failures. The models would not include data from site-specific or generic seismic hazard curves. The designer would determine all significant operational sequences leading to safe shutdown (success paths) using the event trees and fault trees based on fragility data for each component for each success path. The designer would determine the value of the minimum high confidence, low probability of failure (HCLPF) for the plant by determining the HCLPF value for each system. The HCLPF values calculated in this manner is a measure of the robustness of the plant, being an accurate estimate of the earthquake ground motion for which the plant is expected to be able to survive without core damage. If the value of the plant HCLPF is less than about twice the design ground motion zero period acceleration, the designer should perform a more detailed evaluation to find any vulnerability against which to strengthen protection. HCLPF calculations also indicate which components and systems limit the seismic capability of the plant.

ALWR evolutionary designs that include physical separation between safety divisions appear to respond better to internal and external events than do traditional designs. This physical separation reduces the effects of the events and enables the designer to use more deterministic screening methods to assess these effects. Events such as tornadoes and extreme wind may be enveloped using bounding analyses to show that the hazard is insignificant. Bounding analyses of a site-specific external event should either (1) demonstrate that the frequency of occurrence is sufficiently low that it would not significantly contribute to risk at the site or (2) demonstrate that the design would be robust even if the external event occurred. At the design certification, the staff will evaluate fires and internal floods and other external events that are not site dependent.

The Advisory Committee on Reactor Safeguards (ACRS) has concluded that, if enveloping analyses are practical for seismic events and tornadoes, these analyses should be part of the submittal for design certification. The ACRS

agrees that the applicants should submit analyses of other external events that may be site-specific (i.e., storm surge, tsunamis, hurricanes, river flooding and volcanism) for the combined construction/operating license (COL) review. However, the ACRS stated that they are unsure how the staff will resolve any unacceptable findings at the COL stage.

In its letter of May 8, 1992, EPRI stated that the seismic margins assessment is the only suitable margins approach that it knows for evaluating external events. EPRI recommended performing PRA evaluations for those external events that are not eliminated from detailed evaluation.

The staff proposes that ALWR vendors submit analyses of external events beyond the design basis that are separate from analyses required for COL applicants. Different requirements are needed because the site-specific characteristics are not known at the design certification state, and the as-built design details necessary to confirm assumptions made in the vendor's analyses are not all available. Therefore, at the design certification stage, the staff can state its conclusion about the existence of potential vulnerabilities relative to the overall design only. External events analyses sometimes reveal subtle vulnerabilities that can be found only by analyzing detailed designs and by physically walking down the plant.

At the COL stage, the staff will review the site-specific characteristics to ensure that events enveloped by bounding analyses at the design stage have been properly addressed. The staff plans to conduct walkdown inspections to confirm that design commitments have been met.

The staff recommends that the Commission approve the following staff positions regarding analyses of external events beyond the design basis: (1) The staff will require that the analyses submitted in accordance with 10 CFR Section 52.47 include an assessment of internal and external events. PRA insights will be used to support assessment of fire and seismic events. Traditional probabilistic techniques can be used to evaluate internal floods. Site-specific events such as tornadoes and extreme wind may be enveloped using bounding analysis to show that the events are insignificant; and (2) In performing the design certification review, the staff will evaluate external events that are not site-dependent such as fires and internal floods and appropriate bounding analysis. In performing the COL review, the staff will review the site-specific characteristics to ensure that events enveloped by bounding analyses have been properly addressed.

C. Elimination of Operating-Basis Earthquake from Seismic Design

In SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," the staff requested the Commission's approval to separate the magnitude of the ground motion of the operating-basis earthquake (OBE) from that of the safe shutdown earthquake (SSE). The Commission approved the staff's position in its SRM of June 26, 1990. In the draft Commission paper, "Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements," the staff further requested the Commission to

approve eliminating the OBE from the design of systems, structures, and components in both evolutionary and passive advanced reactor designs. The proposed amendment to 10 CFR Part 100 would allow, as an option, that the OBE be eliminated from design certification when the OBE is established at less than or equal one third the SSE. In this manner, the OBE serves the function as an 'inspection level earthquake' below which the effect on the health and safety of the public would be insignificant and above which the licensee would be required to shut down the plant and inspect for damage.

The OBE level that the staff will propose in its revision of Appendix A to 10 CFR Part 100 is one-third of the SSE. This proposed revision states, as an option, that for an OBE equal to one-third or less of the SSE, the OBE can be eliminated from design considerations. The elimination of the OBE from design was requested by EPRI and also recommended by the ACRS in its letter of April 26, 1990.

The staff is assessing the safety margins of several areas of nuclear plant design when the OBE is eliminated from consideration. The industry and staff recognize that earlier seismic criteria resulted in certain aspects of the plant design such as the piping systems being controlled by the OBE. The industry and the staff view the 'controlling' nature of the OBE design as additional margin above the safety margins established by the design bases. Therefore, eliminating the OBE would not result in a significant decrease in the overall plant safety margin. The staff is currently performing a detailed evaluation of the extent to which the OBE controls the design and the effect on the design of structures, systems, and components when the OBE is eliminated from design consideration.

The overall design of reactor site structures is generally conservative, and the structural responses for all combinations of loads, including those from earthquakes, are kept at or below the material yield stresses to preclude plastic deformation.

The staff has examined the structural load combinations and the corresponding acceptance criteria and has determined, on the basis of analyses, tests, and engineering judgment, that the structural design produced by using SSE load combinations envelope those load combinations produced by the OBE. These conclusions are consistent with the staff's licensing experience accumulated during its review of many seismic calculations for individual plants and test data from NRC-sponsored research.

The designer of piping systems considers the effects of primary and secondary stresses and evaluates fatigue caused by repeated cycles of loading. Primary stresses are induced by the inertial effects of vibratory motion. The relative motion of anchor points induces secondary stresses. The repeating seismic stress cycles induce cyclic effects (fatigue).

The staff reviewed these areas and concludes that, for primary stresses, if the OBE is established at one-third of the SSE, the load combinations with the SSE control the piping design when the earthquake contribution dominates the load combination. Therefore, the staff concludes that eliminating the OBE

piping stress load combinations for primary stresses in piping systems will not significantly reduce the existing safety margins because the load combination with the SSE loading is controlling.

Eliminating the OBE will directly affect the current methods used to evaluate the adequacy of cyclic and secondary stress effects in the piping design. Eliminating the OBE from the load combination could cause uncertainty in evaluating the cyclic (fatigue) effects of earthquake-induced motions in piping systems and the relative motion effects of piping anchored to equipment and structures at various elevations because both of the effects are currently evaluated only for OBE loadings.

Accordingly, to account for earthquake cycles in the fatigue analysis of piping systems, the staff proposes to develop guidelines for selecting a number of SSE cycles at a fraction of the peak amplitude of the SSE. These guidelines will provide a level of fatigue design for the piping equivalent to that currently provided in the Standard Review Plan (SRP)(NUREG-0800). Currently, the staff's guidelines in SRP Section 3.9.2 recommend an equivalent of 50 OBE peak cycles for fatigue evaluation. The staff will develop new guidelines after conducting regulatory research and, as necessary, will incorporate the guidelines into an SRP revision or into a regulatory guide. To account for earthquake cycles in the fatigue analyses of piping systems performed until the new guidance is issued, the staff proposes using one-half of the peak amplitude of 75 SSE cycles to evaluate the fatigue of piping. The 75 SSE cycles at one-half its peak amplitude will provide a level of fatigue design for the piping that is equivalent to that currently provided in SRP Section 3.9.2 when the differences in the structural damping between the OBE and SSE are taken into account.

The staff will ensure, as an interim position, that the effects of anchor displacements in the piping caused by an SSE are considered together with the Level D Service limit. The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code), Section III, Paragraph NC/ND-3655 specifies that seismic anchor displacement effects do not need to be considered for Service Level D. The ASME Code requires that seismic anchor motion stresses be considered for Service Level B for which the OBE has been traditionally the designated seismic loading. If the OBE was eliminated from the piping design, the ASME Code, Section III evaluation would have no requirement for considering the effects of seismic anchor motion. The staff's recommendation will correct this deficiency and will require an evaluation of seismic anchor motion effects for the SSE alone. Its effects would be evaluated to the Service Level D limit for which the SSE has been traditionally the designated seismic loading. The staff will continue to develop this approach and consider national consensus codes and standards activities.

Pipe rupture is a rare event that can be caused by errors in the design, construction, or operation; unanticipated loads; or unanticipated corrosive environments. The staff notes that piping failures generally occur at high stress and fatigue locations, such as at the ends of a piping system where it connects to component nozzles.

Recent dynamic pipe tests conducted by EPRI and the NRC demonstrated that the butt-welded piping can withstand seismic inertial loadings higher than an SSE without rupturing. Thus, the staff concludes that the likelihood of a double-ended pipe rupture caused by an OBE level earthquake is remote. Operating experience has shown that pipe failures (splits, through wall cracks and double-ended pipe ruptures) are more likely to occur under conditions caused by normal operation. These conditions include erosion-corrosion, thermal constraint, fatigue, and operational transients.

The staff recommends that, when the OBE is eliminated from the design, no replacement earthquake loading be used to establish the postulated pipe rupture locations. The staff recommends that the criteria for postulating pipe ruptures in high energy piping systems be based on factors attributed to normal and operational transients alone. Further reduction in the number of postulated pipe rupture locations can be considered when compensatory measures are established to minimize the potential for pipe ruptures during normal operating and transient conditions (e.g., control of erosion/corrosion or use of upgrade piping materials). However, the guidelines for the environmental qualification and compartment pressurization are currently based on the mechanistic break locations. Therefore, the staff proposes to revise its criteria for these areas to ensure that each area containing a high energy line be provided with appropriate guidelines to account for the environmental effects of a mass and energy release equivalent to that of a pipe rupture or crack without considering the dynamic effects. The staff plans to develop the environmental guidelines and will issue the guidelines for public comment.

Eliminating the OBE from explicit design consideration affects several aspects of the seismic qualification of safety-related mechanical and electrical equipment. When equipment is qualified by analysis, the acceptance criteria are derived from the ASME Code. Therefore, the conclusions drawn for piping stresses apply equally to this mechanical and electrical equipment and eliminating the OBE should have a negligible effect on the results of analyses performed to qualify equipment. When equipment is qualified by testing, vibration tests are performed for five OBE events with various orientations and then a test for the SSE is conducted. If the OBE is not considered, the equipment may need to be tested first with several fractional SSE loadings and then with one full SSE loading, or the duration of the vibratory motion of the SSE should be changed. Eliminating the OBE will not reduce the margin below the margin provided by current practice. The staff recommends testing equipment to ensure it can withstand fatigue to the degree provided in existing regulatory guidance: 50 cycles at one-half of the SSE peak amplitude or 150 cycles at one-third of the SSE peak amplitude.

The staff will conduct research to determine the number of SSE cycles and the fraction of their peak amplitude for which the equipment is to be tested. The staff will also review the results of research conducted by the industry standards group, as appropriate.

EPRI agrees that the OBE should be eliminated from design and analysis requirements for ALWR designs. The EPRI Requirements Document will require a seismic margins assessment which demonstrates a margin for an earthquake substantially larger than the SSE.

The staff has evaluated the effect on safety of eliminating the OBE from the design load combinations for selected systems, structures, and components. The staff is continuing to review this issue and is developing appropriate criteria for an analysis using only the SSE. The staff is not requesting the Commission to approve the interim positions discussed above. The staff will keep the Commission informed as the review progresses and will note in case-specific safety evaluations, instances in which the applicant proposes to use an SSE-only analysis. The staff will include its recommendations in these reports.

D. Multiple Steam Generator Tube Ruptures

Multiple Steam Generator Tube Ruptures for Passive PWRs

The design-basis accident involving steam generator tube rupture (SGTR) in the current generation of pressurized water reactors (PWRs) is the rupture of one steam generator (SG) tube, with the rate of discharge of primary coolant through the SG tube break greater than the normal charging capacity of the reactor coolant inventory control system. The staff is considering whether multiple SGTRs should be included in the plant design basis for advanced PWR designs. The SRP Section 15.6.3 requires the applicant to analyze for a single SG tube rupture. Currently, there is no requirement to analyze for multiple SG tube ruptures.

In NUREG-0844, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity," of September 1988, the staff estimated the probabilities of single and multiple tube ruptures. When the staff prepared these estimates in 1986, four single SGTRs had occurred in PWRs. All occurred in the United States and all the affected plants were Westinghouse plants: Point Beach Unit 1 (February 1975); Surry Unit 2 (September 1976); Prairie Island Unit 1 (October 1979), and R. E. Ginna (January 1982). Since that time, two more single SGTRs have occurred in the U.S., one at North Anna Unit 1 (July 1987) and another at McGuire Unit 1 (March 1989).

In NUREG-0844, the staff estimated the frequency of a single SGTR to be 1.5×10^{-2} per reactor year (RY). The staff based this estimate on the four events that occurred in approximately 300 "mature" reactor-years (RYs) of operation of Westinghouse plants in the U.S. ("mature" RYs are accumulated after the first 2 years of plant operation). Combustion Engineering and Babcock and Wilcox plants, which had accumulated 77 and 66 mature reactor-years, respectively, at that time without experiencing any SGTRs, were assumed to have the same probability of SGTRs as Westinghouse plants. In the same report, the staff estimated the probability of a multiple tube rupture event, using binomial statistics, as 1.6×10^{-3} per RY. The staff based this estimate on a 50 percent level of confidence (probability) for an event that had never occurred in the approximately 440 mature RY accumulated among all U.S. PWRs at that time.

Since the staff issued NUREG-0844, the total number of mature RY of operation for both Westinghouse PWRs and all U.S. PWRs has approximately doubled. Westinghouse plants have now accumulated approximately 535 mature RYs, and all U.S. PWRs have accumulated about 827 mature RYs. The experience with Westinghouse plants (6 SGTRs in 535 RYs) indicates that the frequency of a single SGTR is approximately 1.1×10^{-2} per RY. Combustion Engineering and Babcock and Wilcox plants appear to have lower SGTR frequencies, since with a failure rate of 1.1×10^{-2} about 3 SGTRs would have been expected in nearly 300 RY of operating experience, and none have occurred. The NRC has not received any report of a multiple SGTR in any U.S. or foreign plant. For consistency with the NUREG-0844 estimate of the probability of multiple SGTR events, a new estimate has been derived based on a 50 percent confidence level for an event that has not occurred in approximately 827 RY of U.S. PWR operation to date. This estimated frequency for a multiple SGTR event is approximately 8.4×10^{-4}/RY.

The causes of SGTRs are generally grouped into two categories: those which occur as initiating events, and those which occur as a consequence of other events that increase the stress on the SG tubes. The probability estimates given above are for SGTRs caused as initiating events. These events include random SGTRs caused by degradation of the tube over time and SGTRs caused by or associated with damage from foreign objects that may be present in the steam generator. Of the 4 SGTRs reported in NUREG-0844, two (Ginna and Prairie Island) are believed to have been caused at least in part by the impact of foreign objects on the steam generator tubes. The SG tubes in other plants have also leaked because of damage from foreign objects, although this leakage did not exceed the makeup capacity of reactor coolant inventory control systems. This issue is of concern in the context of determining the credibility of multiple SGTR events. While it would seem highly improbable that two random SGTR failures would occur simultaneously (as indicated in NUREG-0844), damage or tube failure caused by a foreign object could be a more likely initiator of multiple SGTR. In the Ginna event, the licensee examined the SG tubes after the event and found that although only 1 SG tube had ruptured, more than 20 SG tubes had been severely damaged.

The staff is reviewing the issue of whether to consider a single SGTR or a multiple SGTR as the design basis event for the AP600. The staff is concerned that an AP600 plant could respond in substantially different ways to the two accidents and that a multiple SGTR event could pose substantial challenges to the plant's passive safety systems.

In dealing with a SGTR in a conventional plant, operators isolate the faulted steam generator and reduce the primary system pressure to help stop primary-to-secondary leakage. The operators use the safety-related pressure- and inventory-control systems in these plants (pressurizer spray, high-pressure safety injection) to carry out these procedures. While no multiple tube ruptures have occurred, leak rate would likely increase with the number of tubes ruptured, and the operators would act to mitigate the consequences of the event as quickly as possible. However, the basic procedures to be employed by plant operators in such an event would be similar to those used for single tube ruptures, and the plant conditions would probably be similar

during the transient to those in a single SGTR. Analyses and tests of multiple tube rupture at the SEMISCALE facility have confirmed that the basic plant response is similar to that for a single SGTR event.

The AP600 plant includes no active safety-related inventory or pressure control systems. The core makeup tanks (CMTs) add high-pressure inventory by providing a gravity-driven injection of borated water. A natural circulation passive residual heat removal (RHR) system provides safety-related decay heat removal. The AP600 also uses an automatic depressurization system (ADS) to reduce the primary system pressure in the event of a loss-of-coolant accident (LOCA), which permits injection of a large amount of low-pressure makeup water from the in-containment refueling water storage tank (IRWST). The first stage of the ADS is triggered upon reducing the CMT level to a predetermined point, with subsequent stages actuated as CMT level reaches successively lower values.

Primary system depressurization below that of the secondary system appears undesirable during an SGTR. While the pressure of the primary system should be reduced to about that of the secondary system to inhibit primary-to-secondary leakage, using the ADS will likely further lower the RCS pressure to such an extent that unborated water could flow from the secondary side of the steam generators back into the primary system. This could cause reactivity to increase in the core, with possible detrimental results. Westinghouse representatives assert that the AP600 has been designed with sufficient margin to the ADS initiation setpoint to allow at least 30 minutes of CMT injection after a single SGTR without triggering the ADS. Westinghouse claims that this should be sufficient time for the operators to employ both safety-related and available non safety-related systems to reduce RCS pressure, isolate the faulted steam generator, and terminate the event.

However, if a multiple SGTR occurred, with a substantially greater leak rate of primary coolant, the AP600 may not be able to accommodate the accident without actuating of the ADS. The operators will have substantially less time to bring the event under control before the CMT level is reduced to the ADS setpoint. ADS actuation might result in secondary-to-primary leakage of unborated water. This water could flash to steam as it enters the RCS if the steam generator water is above the saturation temperature at the primary system pressure. Since the AP600's passive safety systems rely on small differential pressures to circulate and inject emergency core coolant (ECC), introducing a large amount of steam into the RCS from flashing secondary water could pressurize the primary system and disrupt or degrade ECC injection. Therefore, contrary to the response of current plants, the plant may respond to a multiple SGTR event in a manner considerably different from that for a single SGTR. The consequences may also differ significantly.

The designer could provide a number of methods for minimizing the consequences of multiple SGTRs, especially to retard or prevent secondary-to-primary leakage or to lessen the amount of reactivity added as a result. These methods include (1) depressurizing the secondary system to maintain the RCS pressure at a value greater than the secondary pressure and prevent back leakage; (2) providing a system to borate the secondary water automatically if it leaks into the RCS from the steam generator; (3) or providing procedures

that inhibit ADS actuation if the primary-to-secondary barrier is breached. The staff is not aware that Westinghouse is considering any of these SGTR mitigation methods for the AP600.

Design basis accidents, such as large-break and small-break LOCAs, have estimated frequencies of occurrence of the order of 10^{-3} per RY to 10^{-5} per RY. These events generally provide the most rigorous test of the plant's safety systems. Recognition that the multiple tube rupture frequency could be in the range of 10^{-3} per RY to 10^{-4} per RY and that the passive plant response for a multiple tube rupture could be significantly different from that for a single tube rupture has led the staff to conclude that rupture of more than a single tube should be considered within the design basis of the plant.

The staff is continuing to evaluate the appropriate number of ruptured SG tubes that should be included within the AP600's design basis. As a minimum, the plant designer or applicant should analyze the multiple SGTRs event to determine, to the extent possible, the quantitative differences in the plant's response to these events. Therefore, as an interim step, the staff intends to require that Westinghouse include analyses of ruptures of 2-5 tubes in its AP600 safety analysis. The staff will report to the Commission when it determines the number of tube ruptures to be incorporated into the design basis of the AP600.

EPRI stated that their Utility Requirements Documents require that future pressurized water reactors have substantially improved capability to handle SGTRs. EPRI also stated that these requirements address improvements in materials, design, and operation to prevent SGTRs and address design features to improve the performance and response of the plant after an SGTR. With regard to multiple SGTR, the EPRI has concluded that the passive plants are not unique with regard to multiple SGTRs, and accordingly, multiple SGTRs should not be included in the design basis for passive LWRs.

<u>The staff recommends that the Commission approve the staff's positions to require that the analysis of multiple SGTRs of 2 to 5 SG tubes be included in the application for design certification for passive PWRs. The staff will evaluate these analyses during the final design approval and design certification review process to help determine the number of SG tube ruptures that will be incorporated into the passive PWR design basis. The staff may require an applicant to submit additional analysis for ruptures of more than five SG tubes.</u>

Containment Bypass Potential Resulting From Multiple SGTRs

The staff has identified an additional containment performance issue that has not been adequately addressed. Specifically a rupture of one or more SG tubes could lead to a bypass of the containment. During a SGTR event, the SG safety or relief valves could be actuated, discharging primary system radioactive inventory outside the containment. The staff concludes that the applicant for design certification should consider providing means to mitigate this containment challenge. This issue applies to both evolutionary and passive PWR designs.

In discussing SECY 90-016 issue III.D on containment performance, the staff emphasized the importance of maintaining containment integrity following a postulated severe accident. The Commission endorsed the staff's goal of reducing the probability for conditional containment failure through use of quantitative guidelines or alternative deterministic objectives. The EPRI Requirements Document states that PWR containments should be designed to provide a leak-tight barrier to prevent uncontrolled release of radioactivity in the event of a postulated (design basis) accident. Containment bypass due to SGTRs could violate containment integrity and hamper meeting the severe accident goals discussed in SECY-90-016.

The staff concludes that containment bypass resulting from multiple SGTRs can be a significant challenge to containment integrity. Therefore, the staff concludes an applicant for design certification or plant designer should consider design features that would reduce or eliminate containment bypass leakage from such a scenario. Features that could mitigate the releases from a tube rupture include:

- A highly reliable (closed loop) steam generator shell side heat removal system that relies on natural circulation and stored water sources.
- A system which returns some of the discharge from the steam generator relief valve back to the primary containment.
- Increased pressure capacity on the steam generator shell side with a corresponding increase in the safety valve setpoints.

The staff recommends that the Commission approve the staff's positions to require that the applicant for design certification for a passive or evolutionary PWR assess design features to mitigate the amount of containment bypass leakage that could result from SG tube ruptures. The applicant or plant designer should consider the mitigation features that would likely be available following a postulated severe accident. The staff concludes that PWR designers should assess such features and address the desirability of this mitigation function. The staff will review this issue when it performs the design certification review.

E. PRA Beyond Design Certification

A plant-specific PRA is an excellent method for assessing overall safety and integrating plant systems and human interactions. Evaluation of a PRA can also reveal important engineering evaluations, assumptions and uncertainties. In the advanced design certification process, PRA insights are used to select among design options, to strengthen the design against previously known vulnerabilities, to characterize the design, and to evaluate the balance between event prevention and mitigation in the design.

At the COL stage, the applicant may be able to provide site-specific information and detailed design information which was not available during the certification process. During the construction stage, the applicant can also consider as-built information. The information from both of these stages could affect the PRA insights previously developed. Experience has shown that subtle design interfaces involving support systems, systems interactions, or man-machine interfaces can significantly affect the risk profile of a plant. The staff concludes that updating the PRA insights can affect implementation of programs in activities such as training, emergency operating procedure development, reliability assurance, maintenance activities, and 10 CFR 50.59 evaluations. Therefore, the PRA should be revised to account for the more detailed information and refinements in the level of design detail. Currently, there are no regulatory requirements that address revising the PRA after it has been completed.

EPRI considers the PRA to be a valuable tool to support plant operation and recommends that the PRA should be revised once the plant is built and maintained for the duration of the operating license.

The staff recommends that the Commission approve the staff's position that, throughout the duration of the combined or operating license, the PRA should be revised to address significant plant modifications, operating experience, and other developments that may affect previous PRA insights. The PRA for each plant could be revised as part of the process for revising the final safety analysis report (FSAR) required in the regulations.

F. Role of the Operator in a Passive Plant Control Room

In SECY 91-272, "Role of Personnel and Advanced Control Rooms in Future Nuclear Power Plants," the staff discussed an issue regarding the role of the operator in a passive plant control room. Specifically, operators in a passive plant control room may use nonsafety-related systems and active "investment protection" systems as the primary means to mitigate transients and accidents (regulatory treatment of these active nonsafety systems is discussed in Section H). Operators will use these systems, before safety-related passive systems are initiated, when responding to transients and accidents. The design of safety-related systems in the passive plant differs significantly from the design of safety-related systems in current operating plants and in evolutionary plant designs.

To operate the passive plant safely, the operator must understand the operation of the "investment protection" systems and their interfaces with the safety-related passive systems. Passive plant operators will be required to perform new functions and tasks unlike those for the evolutionary plants. These new functions and tasks will be due to the new approach in operational philosophy noted above, the increase in automation, and the greater use of advanced technology in the passive plant designs. These new functions and tasks will likely involve greater reliance on monitoring and decision making rather than performing actions directed in procedures. Thus, the design must carefully define the operator's role to ensure that it properly develops the man/machine interface design to facilitate these functions and tasks.

EPRI stated that the ALWR program has provided for "man-in-the-loop" testing during first time engineering specified within Chapter 10 of the EPRI Utility Requirements Document. EPRI indicated that this requirement should adequately ensure that the human component in the man-machine interface system is explicitly included to correct the problem of insufficient focus on the operator in previous designs.

The staff concludes that an extensive man-in-the-loop test and evaluation program will be necessary for the passive plant control room designs. Such testing would likely require a fully functional integrated control room prototype to demonstrate that the passive designs properly consider the operator's role for ensuring plant safety.

Therefore, the staff recommends that the Commission approve the staff's position that sufficient man-in-the-loop testing and evaluation be performed and that a fully functional integrated control room prototype is necessary for passive plant control room designs to demonstrate that functions and tasks are integrated properly into the man/machine interface design. These requirements will be incorporated into the design acceptance criteria (DAC). Each applicant may demonstrate that a control room prototype of reduced scope is sufficient to demonstrate that functions and tasks are integrated properly in the man/machine interface design.

G. Control Room Annunciator (Alarm)[1] Reliability

The annunciator system in a nuclear power plant provides a "first alert" to the control room operator of an abnormal state in the plant, usually over a full spectrum of transients from the malfunctioning of a single piece of equipment to the development of an abnormal state of one or more critical process parameters. The annunciator system also focuses the operator attention on the location and the nature of the malfunction or disturbance. The extent to which this is done is dependent upon the design features of the annunciator system.

Recent events at operating U.S. nuclear plants involving the loss of the plant annunciator system have revealed the vulnerability of the power supply of these systems to single failures. At present, the NRC has no requirements specific to the annunciator system. The acceptance criteria and guidelines for I&C systems important to safety (Appendix A to SRP, Section 7.1) developed from the general design criteria for the I&C, the control room, and the protection and reactivity control systems, do not include the annunciator system by name. IEEE Standard 279, "Criteria for Protection Systems for Nuclear Power Generating Stations," states that the protection systems design should provide the operator with information pertinent to its own status and to the generating station's safety. In a few special cases, specific alarms are required to comply with regulatory requirements because they are essential for the manual initiation of protective actions.

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[1] The annunciator system consists of sets of alarm panels and sound equipment.

When the operating U.S. plants were being designed, the international community observed the same requirements as those in the U.S., with few exceptions such as International Electrotechnical Commission Publication 231A, Supplement to Publication 231, "General Principles of Nuclear Reactor Instrumentation," 1969. This publication gives specific requirements for the design of safety alarms but does not list their functional requirements.

The international requirements changed in 1984 when the International Atomic Energy Agency (IAEA) published Safety Guide D8, "Safety-Related Instrumentation and Control Systems for Nuclear Power Plants." This safety guide discusses top-level requirements for those I&C systems, including the control room annunciator system, that perform functions important to safety but are part of the traditional nonsafety systems.

Safety Guide D8 recommends a method for determining the relative importance to safety and the general principles for developing graded requirements for design features that determine the reliability and availability of these I&C systems. The staff discussed the need for such classification in some detail in SECY-91-292, "Digital Computer Systems for Advanced Light Water Reactors."

The EPRI Utility Requirements Document for both the evolutionary and passive ALWR plants states that

> The main control room (MCR) shall contain compact, redundant operator work stations with multiple display and control devices that provide organized hierarchical access to alarms, displays and controls. Each work station shall have the full capability to perform MCR functions as well as support division of task between two operators.
>
> The display and control features shall be designed to satisfy existing regulations, for example: separation and independence requirements for Class 1E circuits (IEEE Standard 384); criteria for protection systems (IEEE Standard 279); and requirements for manual initiation of protective actions at the systems level (Regulatory Guide 1.62). The M-MIS designer shall use existing defensive measures (e.g., segmentation, fault tolerance, signal validation, self-testing, error checking, supervisory watchdog programs) as appropriate to ensure that alarm, display, and control functions provided by the redundant work stations meet these standards.

Thus, EPRI requires compact workstations with full capability to perform control room functions with fully organized alarms, displays and controls. These workstations, including the alarms, are to be redundant and meet the requirements for independence and separation of Class 1E and associated circuits described in IEEE 384, "Criteria for Independence of Class 1E Equipment and Circuits." The requirements for redundancy also apply to the power supplies associated with these workstations. If the design of redundant stations including the annunciator system does not meet the traditional interpretation of independence (i.e., a master/slave configuration) a single

failure shall not result in a loss of the annunciator system. These requirements form a set of graded requirements for the alarm control, and indication functions that implement the classification approach discussed in SECY-91-292.

The staff concludes that additional requirements for ALWR alarm systems are necessary to minimize the problems experienced by operating nuclear power plants, such as the total loss of power to annunciators because of problems with their power supply. In SECY-91-292, the staff stated that additional requirements could be developed, as needed, from the EPRI Requirements document. The EPRI requirements for redundant control room workstations and displays that include the alarm functions are adequate for these stations.

The staff recommends that the Commission approve the staff's position that the alarm systems for ALWRs meet the EPRI requirements, as discussed above, for redundancy, independence, and separation. In addition, alarms that are provided for manually controlled actions, for which no automatic control is provided, and that are required for the safety systems to accomplish their safety functions shall meet the applicable requirements for Class 1E equipment and circuits.

H. Regulatory Treatment of Nonsafety Systems

Unlike the current generation of LWRs or the evolutionary ALWR designs, the passive ALWR designs rely exclusively on natural forces, such as density differences, gravity, and stored energy for their safety systems to supply safety injection water and provide core and containment cooling. These passive systems do not include pumps, and all valves either require only DC electric power by means of batteries or are air-operated. Check valves will operate by means of pressure differential across the valve. The passive plants will not have safety-related ac electric power. The designers will likely designate all the active systems as nonsafety systems.

Although the passive ALWR designs rely on the passive safety systems to perform the design basis safety functions of replacing reactor coolant and removing decay heat, some of the active nonsafety systems are required to provide defense-in-depth and provide long term recovery capabilities. These active nonsafety systems are the first line of defense during a transient or plant upset, thus reducing challenges to the passive safety systems. These active systems include (1) the chemical and volume control system and control rod drive system, which provide reactor coolant makeup for the passive PWR and passive boiling water reactor (BWR), respectively, (2) the reactor shutdown cooling system and backup feedwater system for removing decay heat in the PWR, and the reactor water cleanup system for removing decay heat in the BWR, (3) the fuel pool cooling and cleanup system for removing decay heat from the spent fuel, and (4) the associated systems and structures to support these functions, including nonsafety-related standby diesel generators. Since these active systems are not designed to meet safety-related criteria, the designers in their Chapter 15 licensing design basis accident (DBA) analyses will not consider these active systems.

Since the passive ALWR design philosophy departs from the current licensing practice, new regulatory and review guidance is needed regarding the scope of staff review on the nonsafety systems. In the draft Commission Paper, "Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements," the staff listed the regulatory treatment of nonsafety systems for the passive design as a significant technical issue and indicated that the staff will propose a resolution of this issue to the Commission. The discussions provided herein are intended to inform the Commission of the staff's continuing effort, outline the staff's proposed approach for reviewing the functional performance of the active nonsafety systems, and seek the Commission's approval of the proposed approach.

In a position paper submitted by a letter of March 19, 1992, EPRI addressed its position on the regulatory treatment of nonsafety systems for passive ALWRs. EPRI's position can be summarized as follows:

1. The safety-related systems, structures, and components (SSCs) and a small group of additional safety features ensure public health and safety consistent with the Commission's Safety Goal Policy Statement. The additional features, which may be nonsafety grade, include those features needed to deal with the anticipated transient without scram (ATWS) and station blackout (codified transients). Requirements imposed by regulation must be addressed by these SSCs and features. PRAs are performed to demonstrate that the specific designs are consistent with these criteria.

2. Augmented protection functions, such as active nonsafety systems, increase the protection for the investor and minimize challenges to safety systems, and therefore are not subject to NRC regulations but satisfy owner requirements established by the Utility Requirements Document (URD). The owners maintain these systems to enhance flexibility for operation and maintenance.

The staff does not agree fully with the EPRI position that nonsafety systems are not subject to NRC requirements. First, although EPRI stated that the passive safety systems (and a small group of additional features) alone will meet the Commission's safety goals, the staff finds that it is very difficult to quantify measure the reliability of the passive systems because of the large uncertainty of passive system performance. The PRA results for the passive plants would likely take significant credit for nonsafety systems. The limited operational experience is not sufficient to address uncertainties with the performance of the passive features and the overall performance of systems to make up reactor coolant inventory and to remove heat from the core and containment.

Although uncertainties associated with the passive system performance can be minimized by carefully planning and performing separate effects and integral system tests, the staff remains concerned about the characteristics of the phenomena associated with the passive systems. For example, the low differential pressures during natural circulation or gravity injection may not provide sufficient means of forcing sticking check valves to operate, unlike the

pumped ECCS which can overcome such stuck valves. The AP600 design relies on the core makeup tanks to provide high pressure passive safety injection. However, the driving head may not be able to inject the CMT fluid into the vessel under all conditions because it may exert less force than the back pressure and flow impedance. The staff is also uncertain about the thermal-hydraulic characteristics of a gravity injection following a depressurization and blowdown to low pressure. Other thermal-hydraulic uncertainties include the effects of fouling on the heat transfer surfaces of heat exchangers and the effects of noncondensibles on natural circulation. These concerns are complicated by the fact that full functional testing of passive systems may not be possible, since core and containment heat removal require that the RCS be depressurized and vented to the containment. It is difficult to confirm the claimed high level of reliability based solely on the passive systems because of their large uncertainties, and therefore, it is important for the active systems to provide diversity and redundancy to the passive systems to prevent core damage.

Second, proposed ALWR requirements specify that the passive systems will be capable of performing their safety functions of cooling the core and containment without operator action or offsite support for 72 hours after the initiating events, and therefore, the design bases of the passive safety systems are centered on the 72-hour capability. After 72 hours, the licensee may need to rely on active systems to mitigate an accident. This increases the need to rely on nonsafety systems for long periods, unlike the practice at current plants, which use active safety systems designed for long-term operation.[2] For example, in SRP 9.5.4, the NRC requires the licensee to store a minimum of 7 days' supply of diesel fuel for each redundant diesel generator system following a loss of offsite power and a design basis accident (DBA). In Regulatory Guide 1.27, the staff specifies that the licensee should provide a safety-related ultimate heat sink with sufficient cooling capacity for at least 30 days to safely shut down the reactor.

Third, the passive DHR designs are restricted by their heat removal rate and the ability of the passive heat removal process, which cannot reduce the temperature of the reactor coolant system below the boiling point of the water, to transfer heat to the in-containment refueling water storage tank (passive PWR design) or the isolation condenser (passive BWR design). The passive DHR systems can only be relied upon to lower primary system temperature to 400°F (after the industry proposed 72 hour accident duration), which would be classified as safe shutdown conditions for the passive ALWRs. The active DHR systems must be relied upon to bring the reactor to cold shutdown. The staff has received limited information on the operation of passive safety systems during reactor shutdown conditions. When a reactor is shut down, the steam generators may be unavailable, and the passive safety injection and heat removal systems may be isolated. Systems that may be isolated when the reactor is shutdown include the accumulators, core makeup tanks, passive safety injection system, and passive RHR system. Therefore, the active

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[2] The concept of long term operations is contained in the SRP and in the Regulatory Guides.

systems may be the only available means of removing heat and making up core coolant. This further emphasizes the importance of active nonsafety systems.

Fourth, the active systems are the first line of defense to prevent challenges to the passive systems. Since traditional safety systems such as auxiliary feedwater and emergency ac power are now proposed to be nonsafety systems, the likelihood of total loss of feedwater or ac power could be higher than it is the current plants. For example, if the nonsafety backup feedwater systems failed in a loss-of-main-feedwater event, the feedwater could be unavailable for an extended period. If the nonsafety standby diesel generators failed during a loss of offsite power, a station blackout could occur. Although the passive safety systems should be designed to mitigate the effects of these events, such events are highly stressful to the operating staff.

Finally, active nonsafety systems and equipment may be important to prevent and mitigate damage to the core and to recover the plant after a severe accident.

For these reasons, the staff concludes that the active nonsafety systems are significant to safety in performing their defense-in-depth functions of preventing and mitigating accidents and core damage. These systems must be considered in evaluating the overall safety of the passive designs. Therefore, the staff does not agree with the EPRI position that these active nonsafety systems are not subject to NRC requirements. Rather, in reviewing the passive ALWR designs, the staff will evaluate both the performance of passive safety systems and the functional capability and availability of the active nonsafety systems to ensure that the plant has a robust defense-in-depth capability to prevent accidents and core damage. The staff desires a high level of assurance that these active systems are available when needed to perform defense-in-depth functions.

The EPRI passive plant requirement document specifies the design requirements for those active systems and equipment that perform defense-in-depth functions. EPRI includes requirements for radiation shielding, redundancy and single active failure considerations, electric power availability, protection against internal flooding and hazards, and system testing capability. For example, the requirements document requires the nonsafety auxiliary systems and components to (1) be arranged and shielded to permit access for operation and maintenance under conditions anticipated during and after nonaccident events leading to operation of the passive safety systems, and meet the requirements for equipment survivability if listed for use as part of severe accident management program; (2) have redundancy to ensure the defense in depth of each system, assuming either (a) a single active failure of equipment which must change state or position to perform the defense-in-depth function or (b) unavailability of this equipment because of maintenance; (3) have available electric power from normal station ac and the redundant equipment (or trains) to be separated to the extent practicable such as by receiving power from separate buses; (4) have redundant components protected against internal flooding, and other requirements related to have included in the design for any hazards in the plant; (5) be demonstrated by testing to be

capable to satisfy the defense-in-depth requirements in system design analyses; and (6) have included in the design for any equipment necessary for plant recovery the capability to operate for the assumed 72-hour accident duration for the expected environment. The EPRI Requirements Document also requires at least one nonsafety system to provide means to maintain the reactor coolant inventory, remove decay heat, and ensure compliance with the specified design limits for infrequent and moderate frequency events without relying on safety systems except for the reactor protection system.

The EPRI requirements are noteworthy and provide important high-level goals for the active nonsafety systems. In particular, EPRI's goal of providing at least one active system that can respond to moderate and infrequent events without actuating the passive safety systems, is an important goal. The staff will review this capability carefully in the passive designs. However, the specific EPRI guidance falls short in specifying requirements to ensure capability and availability of safety-important systems and components. For example, the guidance does not include the scope and degree of required redundancy and separation in sufficient detail and does not specify the methodology for determining the limiting single active failure. EPRI did not specify the requirements for seismic category, wind or external flooding protection, or quality group classification criteria for these systems and components. EPRI also did not specify design criteria (such as IEEE Standard-279) for electrical systems that initiate active systems in the event of a reactor accident and did not specify the inservice inspection and testing requirements. EPRI proposed selection criteria for the technical specification (TS) controls for certain important instrumentation and process variables. EPRI also proposed these criteria for the safety SSCs that are part of the primary success paths and that function or actuate to mitigate a DBA or transient. The active nonsafety systems are not included in the TS control, but can be considered in the justification for continued operation if the passive systems fail to meet the required limiting conditions for operation.

While the active nonsafety systems will not be required to meet strict safety-related requirements, the staff believes that a certain subset or modified subset of the safety-related criteria is required for those important active systems. The performance requirements for an active system should be influenced by the importance of the system to safety, and the designer should consider the safety importance of the active systems during the design process. The staff believes that the active systems should conform to graded requirements based on their importance to safety. Thus, safety important active systems may require more strict criteria than are required for less important systems to ensure appropriate capability and availability. The vendor or applicant should also establish a graded safety classification system for I&C systems as discussed in SECY-91-292. This safety classification system accounts for I&C systems that perform functions that are important to safety, but are not part of the safety system.

The applicant should maintain the operational capability and availability of the important active systems and components by establishing a reliability assurance program that ensures adequate predictive and preventive maintenance

and proper inservice inspection and testing, and TS control that provides appropriate configuration control and system operability requirements during power operation and shutdown conditions.

While formal equipment qualification may not be necessary, it should be demonstrated that these systems will function in the conditions they will experience during plant upset. The staff will evaluate the applicant's proposed system requirements based on safety importance and functional performance requirements.

The staff believes that safety importance of various systems can only be evaluated on a design-specific basis. This is because the relative importance of a system is influenced by the specific system design and arrangements, the test intervals of pumps and valves, and the capability and availability of active systems. The safety importance of systems is highly design dependent, and therefore, the staff requires detailed system design information to perform its evaluation.

To compare the importance of passive safety and active nonsafety systems, the staff, with contractor assistance, conducted an evaluation of a preliminary AP600 design. This was a scoping study because complete design information and success criteria were not available. The staff utilized event sequence diagrams, equivalent to event trees, to define success and failure paths and to find any alternatives to each system included in the design for each event scenario. The staff also used PRA-related methods to measure the importance of the systems in order to compare the importance of the passive systems with that of the active systems. The staff concludes strict criteria are appropriate for those systems found to be of high importance. The staff found that the active nonsafety RHR system had a very high worth, and therefore, may need strict criteria to ensure its functional capability and availability. This system was very important for both LOCA response and decay heat removal during shutdown modes. The staff also found that, while the depressurization system ranked very high in importance, the AP600 included no alternative to this system. Therefore, the CVCS system should meet very strict criteria and there should be a very reliable depressurization system.

The staff concludes that the plant designer should demonstrate that the capability and availability of each system is commensurate with its safety importance. Therefore, to perform the review for design certification, the staff will require the following information on the active nonsafety systems:

1. The plant designer or applicant should perform an evaluation to determine which active systems are important. The safety analysis report (SAR) should include (1) an evaluation of relative safety importance of both passive and active systems (including the methods of analysis, importance measures, accidents and transients scenarios analyzed, and success criteria of the nonsafety systems), (2) an evaluation to determine the limiting single failures, and (3) the preventive and mitigative capabilities of the active systems for severe accidents. The plant designer or applicant should evaluate the required system capability for initiating events from conditions encompassing both power operation and shutdown conditions.

The plant designer or applicant should follow the guidance in the general design criteria (GDC) and current safety standards, such as IEEE 279, to identify important aspects of the design on those active nonsafety systems determined to be important to safety. For example, the GDC require that SSCs important to safety should be designed to meet the quality standards commensurate with the importance of the safety functions to be performed (GDC 1); to withstand the effects of natural phenomena (GDC 2); to minimize the probability and effect of fires and explosion (GDC 3); to accommodate the effects of normal operation, maintenance, testing, and postulated accidents, including LOCA, and to be compatible with the environmental conditions associated with each of these operating conditions (GDC 4); to not to be shared among nuclear power units (GDC 5); and to be provided with an onsite and an offsite electric power system to permit functioning (GDC 17). GDC 34 and 35 require suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities be provided for RHR and emergency core cooling to ensure that the system safety function can be accomplished with onsite or offsite power, assuming single failure. The staff concludes that the plant designer or applicant must, as a minimum, address these GDC and perform an evaluation to show capability and margin. These evaluations should demonstrate that the intent of the GDC have been addressed and provide an explanation if the design deviates from the GDC.

2. The plant designer or applicant should perform thermal-hydraulic analyses to demonstrate adequate capability of the active nonsafety systems to respond to plant upset conditions. The SAR submittal should include (1) a list of those moderate frequency and infrequent events for which the active systems are intended to provide defense in depth by performing preventive and mitigative functions, (2) safety analyses of these events to demonstrate that the nonsafety systems alone meet the ALWR safety design criteria of not challenging the passive safety systems for moderate and infrequent events, and (3) analysis to demonstrate any additional capability to respond to DBA events.

3. The plant designer or applicant should determine the performance and reliability assurance requirements for the important active nonsafety systems to ensure that these systems are operable and available. These requirements may include specific design features for redundancy, separation, quality group, seismic category, protection against fire and flood, the quality assurance (QA) and quality control (QC) program, the reliability assurance program, TS control, maintenance and surveillance, inservice inspection (ISI) and inservice testing (IST), and configuration management activities.

The staff concludes that, with appropriate analysis, active nonsafety systems can be relied upon as a first line of defense-in-depth functions complementing the passive safety systems.

The staff recommends that the Commission approve the staff's approach for resolving the regulatory treatment of the active nonsafety systems. Specifically, a graded approach based on system importance to safety will be used to establish specific requirements of the active nonsafety systems to ensure their capability and availability. This graded approach will require, as a minimum, the plant designer or applicant to submit the evaluations and analyses described above to help verify that the capability and availability of each systems is commensurate with its safety importance. The importance of systems should be evaluated for each design because the specific design and arrangement of systems, and the required test intervals of pumps and valves, and other components will affect the relative importance of various systems. The staff will evaluate each vendor's methodology and criteria used to establish the relative importance of active nonsafety systems, and will evaluate the applicant's proposed system requirements.

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Enclosure 2

ALWR ISSUES CROSS-REFERENCE MATRIX

Category	Issue No.	Issue Title	Commission Papers
I. SECY-90-016 Issues	A.	Use of Physically-Based Source Term	86-228 88-203 89-013 89-153 89-228 89-341 90-016 90-307 90-329 90-341 90-353 92-127
	B.	ATWS	89-153 89-228 90-016 90-353
	C.	Mid-Loop Operation	89-228 90-016 90-353
	D.	Station Blackout	89-013 89-153 89-228 90-016 90-329 90-353
	E.	Fire Protection	89-013 89-228 90-016 90-353
	F.	Intersystem LOCA	89-153 89-228 90-016 90-353

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ALWR ISSUES CROSS-REFERENCE MATRIX (CONTINUED)

Category	Issue No.	Issue Title	Commission Papers
I. SECY-90-016 Issues (cont.)	G.	Hydrogen Control	89-013 89-153 89-228 90-016 90-329 90-353
	H.	Core-Concrete Interaction: Ability to Cool Core Debris	89-153 89-228 90-016 90-353
	I.	High Pressure Core Melt Ejection	89-228 90-016 90-353
	J.	Containment Performance	89-228 90-016 90-353 91-273
	K.	Dedicated Containment Vent Penetration	89-153 89-228 90-016 90-329 90-353
	L.	Equipment Survivability	89-228 90-016 90-353
	M.	Elimination of OBE	89-013 90-016 90-329 90-353 91-135 draft[1]

[1]"Draft" refers to the draft Commission paper, "Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements," which was forwarded to the Commission on February 20, 1992, and made available to the public on February 27, 1992.

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ALWR ISSUES CROSS-REFERENCE MATRIX (CONTINUED)

Category	Issue No.	Issue Title	Commission Papers
I. SECY-90-016 Issues (cont.)	N.	In-Service Testing of Pumps and Valves	89-228 90-016 90-353 91-273

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ALWR ISSUES CROSS-REFERENCE MATRIX (CONTINUED)

Category	Issue No.	Issue Title	Commission Papers
II. Other Evolutionary and Passive Design Issues	A.	Industry Codes and Standards	91-273
	B.	Electrical Distribution	91-078
	C.	Seismic Hazard Curves and Design Parameters	91-135
	D.	Leak-Before-Break	89-013
	E.	Classification of Main Steamline of BWRs	draft
	F.	Tornado Design Basis	draft
	G.	Containment Bypass	draft
	H.	Containment Leak Rate Testing	89-013 89-228 91-273
	I.	Post-Accident Sampling System	draft
	J.	Level of Detail	90-241 90-377
	K.	Prototyping	91-074 91-273
	L.	ITAAC	91-178 91-210
	M.	Reliability Assurance Program	89-013
	N.	Site-Specific PRAs	89-013
	O.	SAMDAs	91-229
	P.	Generic Rulemaking Related to Design Certification	91-262
	Q.	Common Mode Failures in Digital I&C Control Systems	91-292

ALWR ISSUES CROSS-REFERENCE MATRIX (CONTINUED)

Category	Issue No.	Issue Title	Commission Papers
II. Other Evolutionary and Passive Design Issues (cont.)	R.	Beyond Design Basis Analysis of External Events	89-013 draft
	S.	Multiple SG Tube Ruptures	none
	T.	PRA Beyond Design Certification	none
	U.	Control Room Annunciator Reliability	none

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ALWR ISSUES CROSS-REFERENCE MATRIX (CONTINUED)

Category	Issue No.	Issue Title	Commission Papers
III. Passive Design Issues Only	A.	Regulatory Treatment of Nonsafety Systems	89-013 90-406
	B.	Definition of Passive Failure	77-439
	C.	SBWR stability	89-153 91-273
	D.	Safe Shutdown Requirements	draft
	E.	Control Room Habitability	draft
	F.	Radionuclide Attenuation	draft
	G.	Simplification of Offsite Emergency Planning	88-203
	H.	Role of Control Room Operator in Passive Plant	91-272

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Enclosure 3

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9. SPONSORING ORGANIZATION – NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

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10. SUPPLEMENTARY NOTES

Project Number 669

11. ABSTRACT *(200 words or less)*

The staff of the U.S. Nuclear Regulatory Commission has prepared Volume 2 (Parts 1 and 2) of a safety evaluation report (SER), "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document - Evolutionary Plant Designs," to document the results of its review of the Electric Power Research Institute's "Advanced Light Water Reactor Utility Requirements Document." This SER gives the results of the staff's review of Volume II of the Requirements Document for evolutionary plant designs, which consists of 13 chapters and contains utility design requirements for an evolutionary nuclear power plant (approximately 1300 megawatts-electric).

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

advanced light water reactor (ALWR)
Utility Requirements Document (URD)
Electric Power Research Institute (EPRI)
final design approval (FDA)
design certification (DC)
combined operating license (COL)
evolutionary plants
standardization
regulatory stabilization
Advanced Reactor Service Accident Program (ARSAP)
policy issues
optimization subjects
licensability
10 CFR Part 52
Utility Steering Committee

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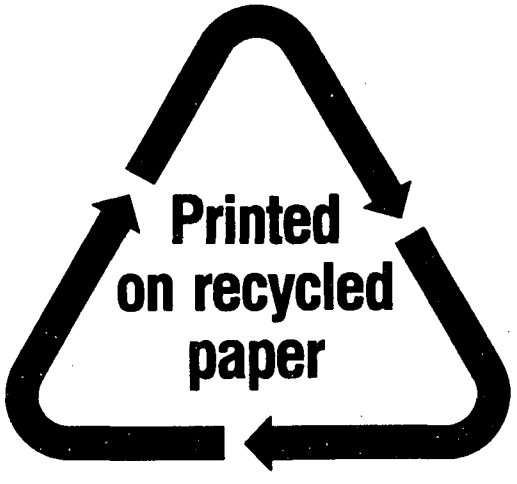

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