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Vol. 3, Pt. 1

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

Passive 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**
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## ABSTRACT

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

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

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

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

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

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

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

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# PREFACE

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

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

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

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

## Outstanding Issues

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

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

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

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

Availability

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

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

# CHAPTER 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 III of the Electric Power Research Institute's (EPRI's) Advanced Light Water Reactor [ALWR] Utility Requirements Document (hereafter referred to as the "passive Requirements Document") through Revision 5. Chapter 1 was prepared, under the project direction of EPRI and the ALWR Utility Steering Committee, by MPR Associates, Inc.; S. Levy Incorporated; TENERA, L.P.; and EPRI.

### 1.1 Review Criteria

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

### 1.2 Scope and Structure of Chapter 1

Chapter 1 of the passive Requirements Document defines the ALWR Utility Steering Committee's overall requirements for the design of nuclear power plants in which passive safety systems will be used for the ultimate safety and protection of the plant.

The key topics addressed in the Chapter 1 review include EPRI-proposed design requirements pertaining to safety, performance, structural design, materials, reliability and availability, constructability, 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 staff requirements memoranda, the Commission instructed the staff to provide an analysis detailing where the staff proposes departure from current regulations or where the staff is substantially supplementing or revising interpretive guidance applied to currently licensed light-water reactors (LWRs). The staff considers these to be policy issues. Annexes A and B of Appendix B to this chapter provide the regulatory analysis of those issues identified as policy issues for the passive plant designs. These issues have been addressed in the Commission paper, SECY-93-087, "Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements," dated April 5, 1993, and in SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety System," dated March 28, 1994.

These policy and technical issues considered fundamental to agency decisions on the acceptability of the ALWR designs are listed below and in Table 5B.1 in Appendix B to this chapter.

- use of physically based source term
- anticipated transients without scram (ATWS)
- midloop 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 (BWR)
- tornado design basis
- containment leak rate testing
- level of detail
- prototyping
- inspections, tests, analyses, and acceptance criteria (ITAAC)
- reliability assurance program
- severe-accident mitigation design alternatives
- generic rulemaking related to final design approval/design certification (FDA/DC)

- regulatory treatment of non-safety systems
- definition of passive failure
- thermal-hydraulic stability of the simplified BWR
- safe shutdown requirements
- control room habitability
- radionuclide attenuation
- simplification of offsite emergency planning

## 1.4 Outstanding Issues

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

Open Issues

P.1.0-1 scope of mitigation requirements (2.1 and 2.4)

P.1.0-2 regulatory treatment of non-safety-related systems (2.3.1, 4.3.1, 7, 9, 12.2.1, 12.2.3, and Appendix B)

P.1.0-3 automatic standby liquid control system for passive BWR design (2.3.2 and Appendix B)

P.1.0-4 check valve categorization (2.3.2)

P.1.0-5 tornado wind speeds (4.5.2)

P.1.0-6 leak before break (4.5.5)

P.1.0-7 seismic evaluation and design of small-bore piping (4.7.3)

P.1.0-8 use of Institute of Electrical and Electronics Engineers 323 (4.8.2)

P.1.0-9 method of environmental qualification of mechanical and electrical equipment (4.8.2)

P.1.0-10 limits on nitrites, nitrates, and total halogens as chlorine (5.2.8)

P.1.0-11 PWR water chemistry (5.5.2)

P.1.0-12 reliability assurance program framework (6.1 and 6.2)

P.1.0-13 quantitative reliability and availability goals (6.2)

P.1.0-14 integration of reliability engineering techniques (6.2, 6.3, and 6.4)

P.1.O-15 relationship of system requirements to overall plant safety reliability and availability goals (6.3)

P.1.O-16 difference between reliability assurance program for safety- and non-safety-related systems (6.2)

P.1.O-17 human factors considerations for operations and maintenance provisions (8.2)

P.1.O-18 computer security reference (11.12)

Confirmatory Issues

P.1.C-1 tornado wind speeds (4.5.2)
P.1.C-2 internal flooding design criteria (4.5.5)
P.1.C-3 compliance with Regulatory Guides (RGs) 1.26 and 1.29 (9)

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

## 1.5 Vendor- or Utility-Specific Items

The DSER for Chapter 1 contained the following vendor- or utility-specific items. The final disposition of each item is discussed in the appropriate section of this chapter, as indicated by parenthetical notation following each item.

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

P.1.V-2 bounding analysis by standard site design parameters (2.3.1)

P.1.V-3 selection of initiating events and their frequency categorization (2.3.2)

P.1.V-4 acceptance criteria for transient and accident analysis (2.3.2)

P.1.V-5 passive plant ATWS response analysis (2.3.2)

P.1.V-6 use of 72-hour design basis (2.3.2)

P.1.V-7 operator actions 72 hours after accident (2.3.3)

P.1.V-8 technical basis for severe-accident management program and emergency operating procedures guidelines (2.3.4)

P.1.V-9 acceptability of analytical codes and methodologies for safety analysis (2.5)

P.1.V-10 defense-in-depth analysis (2.5 and 3.5)

P.1.V-11 60-year plant life (3.3, 4.8.2, 8.2, and 11.3)

P.1.V-12 operation of PWR with a secured reactor coolant pump (3.5)

P.1.V-13 fuel burnup requirements (3.6)

P.1.V-14 extended operating life of control blades and control rod assemblies (3.6)

P.1.V-15 safety classification (4.3.1)

P.1.V-16 seismic qualification by experience (4.3.2 and 4.8.1)

P.1.V-17 non-seismic building structures (4.3.2 and 4.7.2)

P.1.V-18 structural design and construction codes (4.4 and 4.4.1)

P.1.V-19 elimination of operating-basis earthquake from design (4.4.3, 4.7.3, and Appendix B)

P.1.V-20 definition of support group (4.4.3)

P.1.V-21 use of Appendix N of American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section III (4.4.3 and 4.7.3)

P.1.V-22 analysis of vibratory loads with significant high-frequency input (4.4.3)

P.1.V-23 use of nonlinear analysis to account for gaps between pipes and piping supports (4.4.3)

P.1.V-24 probabilistic approach for changing existing loads and/or loading combinations (4.5.1)

P.1.V-25 recurrence interval for wind loadings (4.5.2)

P.1.V-26 maximum ground water level (4.5.2)

P.1.V-27 precipitation for roof design (4.5.2)

P.1.V-28 snow loading (4.5.2)

P.1.V-29 detailed quantification of soil parameters (4.5.2)

P.1.V-30 minimum margin against liquefaction (4.5.2)

P.1.V-31 external hazards evaluation (4.5.2)

P.1.V-32 number of full-stress cycles (4.5.2 and 4.8.1)

P.1.V-33 site-specific safe-shutdown earthquake (SSE) (4.5.2)

P.1.V-34 power spectrum density function of the time history (4.5.2)

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

P.1.V-36 protection against surface vehicle bombs (4.5.3)

P.1.V-37 design against internal-missile generation (4.5.5)

P.1.V-38 design of concrete containment (4.6.1)

P.1.V-39 load combinations for Category I buildings and structures (4.6.1)

P.1.V-40 design of Category I steel structures (4.6.1)

P.1.V-41 combination of pipe rupture loads with seismic loads for seismic Category I structures (4.6.1)

P.1.V-42 combination of loss-of-coolant-accident and SSE loads (4.6.1)

P.1.V-43 load combinations for safety-related portions of the plant (4.6.2)

P.1.V-44 dynamic analysis techniques (4.7.2)

P.1.V-45 methodology for generating design spectra or time histories (4.7.2)

P.1.V-46 structural damping values (4.7.2)

P.1.V-47 masonry walls in Category I buildings (4.7.2)

P.1.V-48 use of expansion anchor bolts - compliance with Office of Inspection and Enforcement Bulletin 79-02 (4.7.2 and 4.7.3)

P.1.V-49 stability of shell-type structures under compression (4.7.2)

P.1.V-50 use of ASME Code Cases N-411 and N-420 in same analysis (4.7.3)

P.1.V-51 use of ASME Code Case N-411 (4.7.3)

P.1.V-52 construction of core support structures (4.7.3)

P.1.V-53 design fatigue curves (4.7.3)

P.1.V-54 use of zinc to reduce radiation fields (5.2.7)

P.1.V-55 grinding controls for PWRs (5.3.1)

P.1.V-56 use of Alloy 600 (5.3.1)

P.1.V-57 effect of fabrication processes on intergranular stress corrosion cracking (5.3.1)

P.1.V-58 selection of seal, gaskets, and protective coatings (5.3.5)

P.1.V-59 aging of cable insulations and other electrical materials (5.3.6)

P.1.V-60 use of hydrogen water chemistry for the advanced BWR design (5.5.2)

P.1.V-61 plant-specific reliability assurance program (6.5)

P.1.V-62 inspection of construction activities (7 and 11.13)

P.1.V-63 installed operating-phase security system (7)

P.1.V-64 reliability of modular construction (7)

P.1.V-65 inspection and verification of security locks robotically (8.3)

P.1.V-66 compliance of FDA/DC applications with Commission's regulations and guidance (10)

P.1.V-67 issue resolution for FDA/DC reviews (10)

P.1.V-68 inspections, tests, analyses, and acceptance criteria (10)

P.1.V-69 implementation of simplification objective (11.4)

P.1.V-70 implementation of standardization objective (11.5)

P.1.V-71 inservice testing requirements for the essential non-safety related components (12.2.1 and 12.2.3)

P.1.V-72 quarterly testing of pumps and valves (12.2.2)

P.1.V-73 check valve testing methods (12.2.2)

P.1.V-74 full-flow testing of check valves (12.2.2)

P.1.V-75 provisions to test hydraulically and pneumatically operated valves under design-basis differential pressure and flow (12.2.2)

P.1.V-76 qualification testing of active and nonactive motor-operated valves (MOVs) (12.2.2)

P.1.V-77 technical concerns regarding MOVs (12.2.2)

P.1.V-78 leak rate testing for individual containment isolation valve (12.2.2)

P.1.V-79 frequency and extent of disassembly and inspection of safety-related pumps (12.2.3)

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

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

P.1.V-2 regulatory treatment of non-safety systems (2.3.1, 2.3.2, and 4.3.1)

P.1.V-3 selection of initiating events and their frequency categorization (2.3.2)

P.1.V-4 licensing-design-basis analysis (2.3.2)

P.1.V-5 transient/accident acceptance criteria (2.3.2)

P.1.V-6 ATWS response analysis (2.3.2)

P.1.V-7 single active failure criteria for check valves (2.3.2)

P.1.V-8 operator actions 72 hours after accident (2.3.2)

P.1.V-9 technical basis for severe-accident management program and emergency operating procedures guidelines (2.3.4)

P.1.V-10 analytical codes and methodologies for safety analysis (2.5)

P.1.V-11 adverse system interactions and effects of non-safety-grade system operation and operator actions (2.5)

P.1.V-12 defense-in-depth analysis (2.5)

P.1.V-13 60-year plant design life (3.3 and 4.8.2)

P.1.V-14 plant response to trips (3.5)

P.1.V-15 fuel burnup requirements (3.6)

P.1.V-16 operating life for control rod blades and control rod assemblies (3.6)

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

P.1.V-18 spectral shifting procedures (4.4.3)

P.1.V-19 analysis of virbratory loads with significant high-frequency input (4.4.3)

P.1.V-20 use of nonlinear analysis to account for gaps between pipes and piping supports (4.4.3)

P.1.V-21 probabilistic approach for changing design loads and combinations (4.5.1)

P.1.V-22 precipitation for roof design (4.5.2)

P.1.V-23 snow loading (4.5.2)

P.1.V-24 effects of wind-induced failure of non-safety structures or safety-related SSCs (4.5.2)

P.1.V-25 analysis or evaluation methods for foundations (4.5.2)

P.1.V-26 number of earthquake cycles (4.5.2)

P.1.V-27 SSE ground motion time history (4.5.2)

P.1.V-28 protection against surface vehicle bombs (4.5.3)

P.1.V-29 leak before break (4.5.5)

P.1.V-30 design against internal-missile generation (4.5.5)

P.1.V-31 design of concrete containment (4.6.1)

P.1.V-32 load combination for Category I buildings and structures (4.6.1)

P.1.V-33 elimination of combination of pipe rupture loads and seismic loads (4.6.1)

P.1.V-34 treatment of loads that reduce the effect of other loads (4.6.1)

P.1.V-35 probability-based mechanistic design loads (4.6.1)

P.1.V-36 compliance with Standard Review Plan (SRP) Section 3.9.3 (4.6.2)

P.1.V-37 dynamic analysis techniques (4.7.2)

P.1.V-38 methodology for generating design spectra or time histories (4.7.2)

P.1.V-39 structural damping valves (4.7.2)

P.1.V-40 masonry walls in Category I buildings (4.7.2)

P.1.V-41 concrete expansion anchor bolts (4.7.2)

P.1.V-42 use of Appendix N of ASME Code, Section III (4.7.3)

P.1.V-43 seismic evaluation and design of small-bore piping (4.7.3)

P.1.V-44 construction of core support structures (4.7.3)

P.1.V-45 pipe support base design (4.7.3)

P.1.V-46 fatigue design curves (4.7.3)

P.1.V-47 use of experience data (4.8.1)

P.1.V-48 justification for use of latest codes and standards (4.8.2)

P.1.V-49 use of zinic to reduce general radiation due to cobalt-60

P.1.V-50 nickel, chromium, and iron alloys (5.3.1)

P.1.V-51 aging of cable insulations and other electrical materials (5.3.6)

P.1.V-52 BWR hydrogen water chemistry (5.5.2)

P.1.V-53 reliability and maintainability analyses (6.2)

P.1.V-54 reliability assurance program organization (6.2)

P.1.V-55 dominant failure modes and risk significance (6.2)

P.1.V-56 plant reliability data system engineering database (6.2)

P.1.V-57 apportionment of contributions to core damage frequency (6.3)

P.1.V-58 NRC construction inspection (7)

P.1.V-59 quality assurance provisions (7 and 9)

P.1.V-60 NRC approval of security system prior to fuel loading (7,11.3)

P.1.V-61 NRC review of modular construction techniques (7)

P.1.V-62 security function details (8.3)

P.1.V-63 compliance with Title 10 of the Code of Federal Regulations (10)

P.1.V-64 plant-specific ITAAC (10)

P.1.V-65 implementation of simplification objective (11.4)

P.1.V-66 implementation of standardization objective (11.5)

P.1.V-67 inservice testing requirements (12.2.1)

P.1.V-68 inservice testing for essential components (12.2.1, 12.2.2, and 12.2.3)

P.1.V-69 check valve testing methods (12.2.2)

P.1.V-70 testing of design-basis capability of safety-related power-operated valves (12.2.2)

P.1.V-71 containment isolation valve testing (12.2.2)

P.1.V-72 frequency and extent of inspection of safety-related pumps (12.2.3)

# 2 SAFETY DESIGN REQUIREMENTS

## 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 accident consequences, 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 passive Requirements Document states that the EPRI-defined SMB contains design requirements that go beyond the minimum required by the Code of Federal Regulations. EPRI states that these design requirements provide safety assurance beyond that provided by the regulatory requirements for investment protection and severe-accident protection.

Section 1.2 of Chapter 1 of the passive Requirements Document states that the primary emphasis of the requirements is on accident prevention, which includes resistance to accidents and prevention of core damage. "Significant additional emphasis" is placed on mitigation. Severe accidents are considered in the design of mitigating systems. The requirements provide design margin by specifying (1) designed-in capability to accommodate transients without challenging engineered safety systems, (2) the operator have significant time to assess and deal with upset conditions with minimum potential for damage, and (3) margin to enhance system and component reliability and to minimize the potential for exceeding limits that might require plant derating or shutdown.

On the basis of the above statements, EPRI states that the ALWR design criteria for passive plants emphasize accident prevention and place "significant additional emphasis" on mitigation. In the DSER for Chapter 1, the staff stated that it was not clear where EPRI placed its "significant additional emphasis" on mitigation. There were many examples throughout the passive Requirements Document about which the staff concluded that the passive plant design criteria placed less emphasis on the mitigation of design-basis accidents (DBAs) than do criteria for current plants. For example, in the DSER for Chapter 5, the staff concluded that with respect to the mitigation issues of control room habitability and fission-product control, the passive Requirements Document provided a lesser level of protective measures than do the current plants. In addition, it was unclear if EPRI's mitigation requirements were intended to include DBAs as well as core-damage accidents. The staff concluded that EPRI should clarify these statements. Mitigation of accidents is discussed further in Section 2.4 of this chapter. This was identified as an open issue. In a letter dated March 31, 1993, EPRI provided a marked-up page to revise Section 1.2 of the passive Requirements Document to clarify that significant additional emphasis is placed on mitigation of severe accidents. The staff concludes that this change is acceptable.

## 2.2 Accident-Resistance Requirements

Section 2.2 of Chapter 1 of the passive 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; lower core power densities; a larger reactor vessel, pressurizer, and steam generator secondary side; a longer transient response time; 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 passive 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) prevent operator override of safety system functions as long as a valid safety system actuation signal exists; (4) require no safety-grade onsite ac power generator except for dc-converted ac power for instrumentation and control functions; and (5) have the capability of achieving safe shutdown with safety-grade equipment only, assuming the most limiting single failure.

These requirements are generally acceptable. However, the requirement regarding a non-safety-related onsite ac power source affects the reliability and availability of those non-safety-related active systems that provide defense-in-depth functions. This issue is enveloped for the passive designs under the issue pertaining to the regulatory treatment of non-safety systems (see the regulatory departure analysis in Appendix B to this chapter). In the DSER for Chapter 1, the staff identified this regulatory treatment of non-safety systems (RTNSS) in the passive plant designs as an open issue. As discussed in Section 2.2 of Chapter 3 of this report, EPRI in its letter of May 26, 1993, provided a structured process for the evaluation of the RTNSS in the passive plant designs. In Revision 5, EPRI provided its requirements for implementing the RTNSS process in Section 1.6 of Appendix A to Chapter 1 of the passive Requirements Document. This process will be used by the plant designer to identify risk-significant structures, systems, and components, as well as their reliability/availability missions and regulatory oversight

measures. The staff has found this RTNSS process to be acceptable and will ensure that passive plant vendors follow it. It will evaluate the issue of non-safety-related onsite ac power source for active systems during its review of an individual application for FDA/DC. Therefore, this open issue is closed.

Physical Security

Section 2.3.1.4 of Chapter 1 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 Section 73.55(d)(7)(ii) of Title 10 of the Code of Federal Regulations (10 CFR) 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 2.3.1.9 of Chapter 1 addresses plant siting. Table 1.2-6 gives 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 passive 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 modification of 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 of the Requirements Document for evolutionary plant designs, 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 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. Section 2.3.2.3 requires that the initiating events be classified by the frequency-of-occurrence categories and that events in each category by themselves cannot generate more serious incidents without other incidents occurring independently. Sections 2.3.2.4 and 2.3.2.5, respectively, specify 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 2.3.2.8 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 passive Requirements Document is based on that of RG 1.70, "Standard Format and Content of Safety Analysis Reports 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, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," 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, "Accident Analysis," 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 passive Requirements Document is, therefore, consistent with the Commission's current licensing approach.

The events and accidents in Table 1.2-1 of Chapter 1 are generally consistent with the initiating events described in Chapter 15 of the SRP with the following exceptions. Item 3.1, "Reactor coolant pump trips (PWR)," should include both partial and complete reactor coolant pump trips (both are moderate-frequency events). Category 2, "decrease in heat removal by secondary system," should include steam pressure regulator failure and turbine trip without bypass as moderate-frequency events. Item 6.1, "inadvertent opening of a pressurizer safety valve in a PWR or a safety or relief valve in a BWR," should continue to be categorized as a moderate-frequency event rather than an infrequent event. There are also certain acceptable exceptions because of the design features unique to the ALWR passive plants. For example, the recirculation pump trip and pump shaft seizure events for BWRs are not included in the list because there are no recirculation pumps in the passive BWRs. Control rod assembly malfunctions at passive BWRs may also be excluded if proper justification can be provided for excluding them because of unique design features, interlocks, and routine surveillance. However, because the passive Requirements Document does not present an actual design, the staff concludes that there is insufficient justification to delete consideration of the rod drop event as an accident. The plant designer must justify the

exclusion of BWR control rod assembly malfunctions from consideration in safety analyses during the design-specific review.

By letter dated September 29, 1992, EPRI stated that Table 1.2-1 was not an inclusive listing of events to be considered in the safety analysis. The staff agrees that Table 1.2-1 is not inclusive and will require the designer to perform a safety analysis for a complete set of transients and accidents relevant to the specific passive ALWR design. Because selection of the appropriate set of initiating events and their frequency categorization is design specific, the staff will evaluate the completeness of the selected transients and accidents during its review of an individual application for FDA/DC by considering unique design features. This list of transients and accidents should not be limited to those that have been historically analyzed. The selection of transients and accidents and their respective event frequency categories must be documented and justified for each ALWR design. Deviations from RG 1.70 and the event identification or frequency categorization in the SRP must be adequately justified by the vendor.

Sections 4.55 through 4.60 of Appendix B to Chapter 1 of the passive Requirements Document identify several items that comply with the intent of the following SRP sections regarding transient and accident analyses:

- 15.1.5, "Steam System Piping Failure Inside and Outside of Containment (PWR)"
- 15.2.6, "Loss of Nonemergency AC Power to the Station Auxiliaries"
- 15.2.7, "Loss of Normal Feedwater Flow"
- 15.2.8, "Feedwater System Breaks Inside and Outside Containment (PWR)"
- 15.5.1-2, "Inadvertent Operation of ECCS [Emergency Core Cooling System] and Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory"
- 15.6.5, "Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary"

The passive Requirements Document states that the SRP was prepared presuming plant designs generally representative of today's LWR designs. The passive ALWR design is significantly different from current LWR designs in that the core cooling systems are passive systems that will function without the need for ac power. Although the specific assumptions regarding the evaluation of these events are not consistent with that of the SRP, the intent of the SRP is satisfied. Because the above events are identified in Table 1.2-1 of Chapter 1 for LDB safety analysis, the staff agrees that the passive plant design criteria meet the intent of the SRP. However, the staff will evaluate the LDB analysis during its review of an individual application for FDA/DC.

Transient/Accident Acceptance Criteria

Table 1.2-2a of Chapter 1 of the passive Requirements Document specifies the acceptable reactor and fuel design limits and consequence analysis limits for moderate-frequency, infrequent, and limiting fault events. In Revision 2 of

the passive 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). These criteria are consistent with current licensing practice and are, therefore, acceptable.

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 2 of the passive 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 criteria will be less than the 95/95 departure from nucleate boiling ratio (DNBR) limit, and for the BWR, it will be less than the minimum critical power ratio (MCPR), 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 will require each plant designer to perform an analysis for the ATWS events to demonstrate that passive plant ATWS response is consistent with that considered by the staff in its formulation of the 10 CFR 50.62 design requirements for current plants. The staff will address this matter during its review of an individual application for FDA/DC.

In the DSER for Chapter 1, the staff stated that 10 CFR 50.62 requires each BWR to have a standby liquid control system (SLCS) that is automatically initiated, whereas Section 2.5.5 of Appendix B to Chapter 1 of the passive 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 Requirements Document for evolutionary plant designs to reflect that position. In the DSER for Chapter 1, the staff stated that

although EPRI had not submitted its position regarding the automatic actuation of the SLCS for passive designs, the staff expected EPRI to provide design requirements that were consistent with those for the evolutionary designs regarding this matter. Until then, the staff considered this to be an open issue. In Revision 4 of the passive Requirements Document, Section 2.5.5 of Appendix B to Chapter 1, Sections 4.5.1.2, 4.5.2.4, and 4.5.3.3.1 of Chapter 5, and Section 8.7.2 of Chapter 10 were revised to require an automatic initiation of the SLCS. This is consistent with the requirement of 10 CFR 50.62 for ATWS and is acceptable. Therefore, this open issue is closed.

Check Valve Categorization

Section 2.3.2.5 of Chapter 1 of the passive Requirements Document specifies that the plant designer will identify potential single equipment failures that could occur at the same time as initiating events. The types of single failures to be considered include failures of active components and electric equipment, operator error, and passive failures occurring 24 hours after the initiating event. Tables 1.2-1a and 1.2-1b of Chapter 1 provide examples of ALWR coincident single failures for BWRs and PWRs, respectively. Even though the examples of single failures in these tables are extensive (including single failure consideration for valves in passive safety systems such as the PWR core makeup tank, accumulators, and makeup lines of the in-containment refueling water storage tank, and the BWR gravity-driven core cooling line), no specific mention is made of check valves being a single failure consideration.

The passive Requirements Document was inconsistent in its categorization of check valves as active or passive components. Section 2.2 of Chapter 1 categorizes check valves and noncycling powered valves as passive components. Section 4.2.3.1.1 of Chapter 5 originally indicated that check valves were active features. However, in Revision 4 of the passive Requirements Document, check valves as active features were removed from this section.

There is no clear-cut definition of whether a failure of a check valve to move to its desired position is an active or passive failure. American National Standards Institute/American Nuclear Society (ANSI/ANS) 58.9-1981 cites the failure of a check valve to move to its correct position as an active failure. SECY-77-439 indicates failure of a check valve to move to its correct position when required as a passive failure. An International Atomic Energy Agency (IAEA) paper entitled "Application of the Single Failure Criterion - A Safety Practice" (Safety Series 50-P-1) states that in some Member States a failure of a simple swing-type check valve to open need not be considered as a single failure, whereas in other Member States self-operating components such as check valves are considered to be active components if the state of the component is changed during the given event sequence after a initiating event. Considering the test intervals of check valves of about 1 year, the IAEA paper determines that the probabilities of failure of check valves to open or close are closer to the failure probabilities of active components than those of passive components. The IAEA paper also states that a conservative approach is to assume a check valve failure in the single failure analysis.

For current plants, the staff practice has normally been to treat check valves, except for containment isolation systems, as passive devices during transients or design-basis accidents and, therefore, not subject to single

active failure consideration. In the DSER for Chapter 1, however, the staff stated that it was considering redefining the failure of check valve to that of an active failure and identified this as an open issue. This consideration is necessary because safety-related check valves in the passive designs will operate under different conditions (low flow and pressure differential without pump discharge pressure to open valves) from the current-generation reactors and the evolutionary designs. In addition, check valves have high safety significance in the operation of the passive safety systems, and operating experience of check valves shows that they may have a lower reliability than originally anticipated.

Although Section 2.2 of Chapter 1 of the passive Requirements Document categorizes check valves as passive components, EPRI in its August 1, 1991, response to a staff question, also stated that the ALWR program endorses ANSI/ANS 59.9-1981 considering check valves to be active components when they are required to change state to perform their safety function. Failures of these components are considered to be active failures that occur coincident with event initiators. It is also noted that the ANSI/ANS standard allows exemptions where the proper function of a component can be demonstrated despite any credible condition, and it requires documentation of the exemptions in the single failure analysis. EPRI further stated in the letter that check valve applications in the passive safety systems would be reviewed thoroughly under the ALWR reliability program. This will include identifying the particular check valves that play a key role in ensuring that the core damage frequency requirements are met, reviewing whether available check valve reliability data are applicable and sufficient for passive plant safety systems, and identifying appropriate measures for ensuring continued check valve reliability during plant operating life. EPRI indicated that the intent of its work is to use a rigorous evaluation to develop the best technical solution rather than simply relying on single failure to ensure safety. In a position paper, "NRC Policy Issue Analysis and Recommendation," contained in a May 5, 1992, letter, EPRI contended the passive Requirements Document requirements, ALWR safety goals, and the iterative use of probabilistic risk assessment (PRA) in the design process ensure that the unavailability of check valves will be sufficiently low and independent of the initiating failure that check valves need not be assumed to fail. Thus, EPRI recommended that check valve failures not be redefined as active failures for the passive safety systems. In its letter of December 10, 1992, however, EPRI also stated that this industry position is consistent with ANS 58.9.

Consistent with ANS 58.9, the staff position is that, except for those check valves whose proper functions can be demonstrated and documented, check valves in the passive safety system designs will be subject to single active failure consideration. In determining an exemption to single failure consideration for a particular check valve application the plant designer must perform a comprehensive evaluation of appropriate check valve test data and operational data of similar check valve designs in similar applications, and operating environments to demonstrate that the reliability of the particular check valve application is such that the failure probability is comparable to those of passive components. An example of potential exemption is the accumulator check valves whose applications are identical to those for currently licensed plants where the accumulator pressure will usually provide large pressure differential to force open the valves as the reactor coolant system pressure falls. The staff recommended this position for the Commission's approval in

SECY-94-084. In its staff requirements memorandum (SRM) dated June 30, 1994, the Commission approved the staff's recommendation. The SRM continued that in some situation, a design that consider such failures may overall be less reliable (due to added complexity and new failure modes) than one when the valve is treated as passive.

The staff concludes that the passive Requirements Document does not have sufficient information to justify exemption of the single failure consideration a priori for all check valves in the passive ALWR designs. During its review of an individual application for FDA/DC, the staff will evaluate justifications for those check valves that are not designed to meet the single active failure criteria. Therefore, this open issue is closed.

Seventy-Two-Hour Timeframe Design Basis

One of the principal design requirements of the passive Requirements Document is that the core must be cooled and containment integrity maintained for 72 hours without reliance on ac power and operator action. Section 2.3.2.9 of Chapter 1 specifies that no operator action will be necessary to meet regulatory limits until at least 72 hours following the initiating licensing design-basis events, and Section 1.2.1.1 of Chapter 5 specifies that the passive systems should be able to perform their safety functions independent of operator action or offsite support for 72 hours after the initiating event.

The design bases of the passive safety systems specified in Chapter 5 of the passive Requirements Document are centered on the 72-hour capability. For example, Sections 2.2.10 and 2.2.11 originally specified that passive engineered safety systems equipment will be designed and qualified for operation as required in the environment expected for 72 hours after an assumed accident and coincident loss of ac power, and will not require support functions from non-safety-related systems during this period. Section 3.2.2 of Chapter 5 states that the passive safety injection system core coolant inventory should be adequate for at least 72 hours without the need for operator action. Section 4.3.3.5 requires that the passive decay heat removal (PDHR) systems have sufficient water capacity in the PDHR water pool to permit 72 hours of operation after scram without the need for refill.

Not only is the 72-hour timeframe used as the design basis for the safety systems, it is also used as a justification for not requiring support systems to be designed for safety-related functions. These design bases are consistent with the policies for passive ALWRs stated in Volume I of the Requirements Document. However, the passive Requirements Document does not provide sufficient justifications or bases to substantiate these policies and objectives. Section 2.4.2.8 of Chapter 5 of the passive Requirements Document specified that non-safety-related equipment necessary for plant recovery after the assumed 72-hour accident duration will be designed for the expected environment during the 72-hour period. However, the passive Requirements Document did not identify which equipment will be needed for (1) plant recovery after 72 hours and (2) continued accident mitigation. Although Chapter 3 of the passive Requirements Document specifies that the active support systems will provide defense-in-depth functions, such as reactor coolant makeup and decay heat removal functions, they are not required to meet the requirements for safety-grade systems. In addition, the passive Requirements Document specifies that only simple, unambiguous operator actions and

easily accomplished offsite assistance will be necessary after 72 hours to prevent fuel damage. However, there is no clear definition of what constitutes "simple, unambiguous operator action and easily accomplished offsite assistance." In Section 1.2.1.3 of Chapter 5 of the passive Requirements Document, EPRI stated that the passive plant control room will be designed so that it can be maintained during the 72-hour design period as the primary location from which personnel can safely operate in case of an accident. In Section 2.1.1 of the DSER for Chapter 5, the staff concluded that the "72-hour design-basis period" was not consistent with general design criterion (GDC) 19 of Appendix A to 10 CFR Part 50. The staff's position was that the duration of the accident, as specified in GDC 19, should be used as the design-basis period, rather than 72 hours, for control room habitability assessment.

Since the safety systems were originally designed for 72-hour duration after the initiation of accidents and may rely on other non-safety-related active systems to perform core damage prevention and mitigation functions thereafter, the staff was concerned about the reliability and availability of these non-safety-related active systems. Also, RG 1.27, "Ultimate Heat Sink for Nuclear Power Plants," requires that a safety-related ultimate heat sink consisting of at least two sources of water be provided with the capability to provide sufficient cooling for at least 30 days to permit simultaneous safe-shutdown and cooldown of all nuclear units that it serves and maintain them in a safe-shutdown condition. Since the PDHR systems provide a safety-grade means to transfer decay heat to the ultimate heat sinks, the requirement of a 72-hour capability for these systems is inconsistent with the 30-day capability of safety-related ultimate heat sink specified in the regulatory guide. In the DSER for Chapter 1, the staff identified the design basis of 72-hour capability of the passive safety systems as a vendor-specific item and stated that each applicant must justify the use of the 72-hour timeframe design basis, the type of equipment and its quality and availability for plant recovery after 72 hours, and demonstrate that available non-safety systems will provide for long-term cooling.

EPRI has revised the passive Requirements Document requirements related to the 72-hour capability. In Revision 3, Section 2.2.10 of Chapter 5 was revised to further require that the passive safety systems continue to be available for an extended period of time to perform their safety function by using onsite facilities and by connecting readily available offsite equipment and resources to replenish these inventories and provide support as may be necessary. The onsite facilities and the connections to utilize offsite support will be designed to be readily accessible and to withstand SSE conditions. Offsite equipment and resources required to maintain the plant in a safe state must be shown to be readily available from normal commercial sources or from a pre-arranged source not subject to the same external event that may occur at the site, and to be placed in readiness at the plant within the initial 72-hour period, as need. Section 2.2.11 of Chapter 5 was revised to require plant designers to define the recovery process to achieve cold shutdown. It requires the passive safety systems equipment to be environmentally qualified for an extended period until the recovery is ensured, and have the capability to use non-safety systems for the recovery operation if the passive system operation is terminated prematurely. Section 2.4.2.8 of Chapter 5 was revised to require that non-safety-related equipment necessary for plant recovery after the assumed 72-hour accident duration be designed for the expected environment in the extended period.

In Revision 3 and 4, Section 6.4.5 of Chapter 5 was revised to require (1) a passive, safety-grade control room pressurization system that would use bottled air to keep operator doses within the limit of GDC 19 and SRP Section 6.4 for the first 72 hours of an accident and (2) safety-grade connections for the pressurization systems to allow the use of offsite, portable air supplies after 72 hours to minimize operator doses. The staff's evaluation of this requirement is given in Section 6.5 of Chapter 5 of this report.

The staff concludes that these requirements relative to the 72-hour capability are acceptable, provided the reliability/availability (R/A) of these onsite and offsite supporting equipment and resources are such that the safety functions of the passive systems are ensured for the time needed. As discussed in Section 2.2 of Chapter 3 of this report, EPRI in its May 26, 1993, letter provides a structured process for evaluating RTNSS in the passive plant designs. This RTNSS process will be used by plant designers for determining the risk significance and R/A missions of non-safety systems. Appropriate regulatory oversight of these systems, such as reliability assurance program, administrative control, and technical specifications, will be developed based on their risk significance and R/A missions. The 72-hour design basis for passive safety system capability and operator actions 72 hours after accident initiation will be evaluated as an integral part of the RTNSS process. Therefore, the appropriate R/A and regulatory oversight of the on-site and offsite supports needed to replenish the passive safety systems inventories for extended operation will be evaluated and determined. The staff will verify the appropriateness of this evaluation and results during its review of individual applications for FDA/DC. The staff's evaluation of EPRI's requirements for capability and reliability of the non-safety systems to enable the plant to achieve cold shutdown is presented in Section 3.3 of Chapter 5 of this report.

### 2.3.3 Safety-Margin-Basis Requirements

Section 2.3.3 of Chapter 1 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 best-estimate analysis.

Section 2.3.3.2 of Chapter 1 states that the plant will be designed so that only simple, unambiguous operator actions and easily accomplished offsite assistance will be necessary 72 hours after the initiating event to prevent fuel damage. It also requires that the operator have 30 minutes or more to act to prevent (1) damage to equipment or plant conditions resulting in a significant outage and (2) core damage during potential core-damage accident sequences during which manual action is necessary. The staff concludes that each application for FDA/DC must clearly define the "unambiguous" and "easily accomplished" operator action and the operator response time, that is, the time after an alarm or other means provide clear indication of the event sequence requiring operator actions.

Section 2.3.3.7 of Chapter 1 specifies that the probabistic risk assessment (PRA) analysis, 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 passive 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 components is maintained; (2) develop the technical basis for a severe-accident management program to ensure core-damage prevention and mitigation; 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 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 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 implementation of 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 judgements 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, the staff will not use this goal as a staff acceptance criterion.

2.3.4 Emergency Operating Procedures

Section 2.3.1.6 of Chapter 1 of the passive Requirements Document specifies that the passive ALWR plant will be designed to prevent operator override of safety system functions as long as a valid safety system actuation signal exists. Sections 2.3.2.9 and 2.3.3.2 of Chapter 1 specify that no operator action will be necessary until at least 72 hours following the initiating events, and only simple, unambiguous operator actions and easily accomplished offsite assistance will be necessary after 72 hours to prevent fuel damage. Section 2.3.3.9 of Chapter 1 requires the plant designer to develop the technical basis for a severe-accident management program, including emergency procedure guidelines, to ensure core-damage prevention and mitigation, including meeting the offsite dose limits. Section 2.2.4 of Chapter 5 specifies that the engineered safety systems design will be simplified to make the operators' procedures less complex and interdependent and to reduce the need for protective interlocks.

Although the plant is required to be designed to prevent operator override of safety system functions, EPRI also specifies in Section 2.2.13.1 of Chapter 1 that all safety systems will be provided with the capability for manual as well as automatic actuation, and be designed to ensure successful completion of their safety functions independent of any operator control actions. Therefore, the plant design features will limit the operators' ability to override safety signals, but will not unduly restrict their ability to respond to system failures or unanticipated event progression.

Since the passive plant will be designed to require no operator actions for 72 hours after event initiation, the staff was concerned about the potential for the operators to commit human errors of commission that might disrupt the safety functions of the passive systems. This could result from a potential inadequacy of the emergency operating procedures (EOPs), which either fail to prevent, or even require, unwanted operator actions during an accident. In its letter dated May 13, 1991, EPRI stated that, as part of the development of the technical bases for EOPs, it is possible that a limited number of actions may be identified that lengthen the time for equipment recovery or that minimize the magnitude of postulated challenges to the fuel, reactor, or containment that require the inhibiting of safety systems actuation. If such actions were identified, analysis of the plant conditions that benefit from these operator actions would be required as part of EOP development to ascertain (1) the symptoms leading to the need for the action, (2) the time available for accomplishing the action, (3) the task analysis to ensure the action can be accomplished in the environment and the time available to the operator, and (4) the availability of instrumentation to provide unambiguous indication of the symptoms.

EPRI also stated that the EOPs (or guidelines) will be provided by the plant designer as an integral part of the total plant design, will be based on detailed analyses of the functions and tasks of the operators, and will be verified and validated by active simulation. If there are situations in which there is the potential for a human error of commission, the operators will be provided with procedural guidance and, if that is not sufficient, physical interlocks and inhibits will be incorporated in the design. Also, the procedures will not preclude operator action after the initiation of a passive system if that action can mitigate the consequences of an event; however, any such actions will have to be consistent with the need to permit safety system actuation to proceed to completion. Furthermore, such procedural guidance will have to be fully verified and validated.

In addition, the normal plant design configuration limiting the ability to initiate overrides, along with the inclusion of unambiguous symptoms for manual control of safety systems, will limit the contribution of errors of commission to plant risk.

The staff concludes that these design objectives are acceptable. However, it will evaluate the technical bases for the severe-accident management and EOP guidelines, and the design of acceptable means (i.e., hardware or software inhibits) to prevent operator override of safety system functions without unduly restricting the operators' ability to respond to system failures or unanticipated event progression during its review of an individual application for FDA/DC.

Although Section 2.3.3.7 of Chapter 1 of the passive Requirements Document excludes sabotage events from the PRA, deterministic analysis of the vulnerability to sabotage is required in Section 5.2.2.1 of Chapter 9 of the passive Requirements Document. In its letter dated March 1, 1991, the staff asked EPRI to discuss if this sabotage vulnerability analysis should be specified as additional information relevant to the development of the emergency procedure guidelines and severe-accident management program. In its letter dated May 13, 1991, EPRI noted that Section 2.3.3.9 of Chapter 1 of the passive Requirements Document requires the plant designer to use not just the PRA but

any other relevant information. The staff concludes that the term "relevant information" is broad enough to encompass procedure guidance suggested by the vulnerability analyses.

## 2.4 Mitigation Requirements

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

In the DSER for Chapter 1, the staff stated that Section 2.4.1 of Chapter 1 provided 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 did not include requirements for fission-product control or hydrogen control during design-basis loss-of-coolant accidents. The staff stated that EPRI should either include these two issues or justify their exclusion. This was identified as an open issue. In Revision 5, EPRI revised Section 2.4 of the passive Requirements Document to include fission-product control and hydrogen control during design-basis accidents in the mitigation requirements. The staff finds this change acceptable. Therefore, this open issue is closed. The staff discusses these issues further in Appendix B to this chapter.

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, emergency planning, and hydrogen control.

## 2.5 Analysis Requirements and Acceptance Criteria

Section 2.5 of Chapter 1 of the passive Requirements Document specifies the EPRI-proposed requirements for the licensing-design-basis (LDB) and safety-margin-basis (SMB) analyses. EPRI states that for the LDB analysis, NRC-approved methods will be used including assumptions and limits that are based on actual physical conditions during the accident or transient being analyzed with conservatism consistent with regulatory 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. In Revision 5, EPRI added Section 2.5.1.4, stipulating that the requirements in Section 1.6 of Appendix A to Chapter 1 of the passive Requirements Document for implementing the RTNSS process must be met. Section 2.5.2.5 states that for events considered in the licensing 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 passive 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. Acceptable analytical codes used for ALWR analyses must have the capability to handle system performance and phenomena unique to passive ALWR designs, such as operation of gravity-fed safety injection systems, natural circulation heat exchangers at both high and low pressures and in the presence of noncondensible gases, critical flow in the automatic depressurization valves over the range of expected operating conditions, boron transport under gravity-driven system conditions, and natural circulation stability of the simplified BWR design. The codes must also be capable of handling interactions among passive safety systems as well as active systems. The analytical methods and codes must be validated with appropriate test data, including separate effects and integral effect tests with consideration of scaling effects. In its letter dated July 1, 1991, regarding the need for large-scale, full-height, and full-pressure integral testing of the passive safety systems, EPRI indicated that the ALWR Utility Steering Committee has established an analysis and testing review team, which is reviewing the detailed analysis and test plans for the passive plant designs and providing appropriate guidance. 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.

During accident conditions, operator actions and the operation of non-safety-grade active systems could result in plant conditions different from or more severe than those assumed in the safety analysis. Therefore, although credit for non-safety-grade equipment is not normally allowed for design-basis accidents, the adverse system interactions and effects of non-safety-grade system operation and operator actions must be examined and, if they exist, accounted for in the safety analysis. The staff will address this during its review of an individual application for FDA/DC.

Section 2.5.3 of Chapter 1 requires the plant designer to define the requirements and acceptance criteria for the SMB analyses, which will be based on best-estimate models. EPRI states that since the SMB is outside the scope of the LDB, the plant owner rather than the NRC should specify the requirements for investment protection. Because of the defense-in-depth philosophy of using non-safety-grade active systems to prevent unnecessary actuation of the passive safety systems. Section 2.3.2.1 of Chapter 3 of the passive Requirements Document states that a non-safety-grade auxiliary system will ensure that specified design limits for the plant's infrequent and moderate-frequency events are not exceeded without reliance on safety systems except the reactor protection system. Section 2.3.3.5 of Chapter 3 also requires analyses of the non-safety system designs be performed to demonstrate their system capability to satisfy this defense-in-depth requirements of Section 2.3.1 of Chapter 3. The staff will review these analyses to verify that defense-in-depth capability exists during its review of an individual application for FDA/DC. The staff agrees that best-estimate evaluations, rather than more conservative LDB analyses are appropriate. The reliability requirements for the defense-in-depth active systems are addressed in Chapter 3 of this report.

## 2.6 Criteria and Methodology for ALWR Emergency Planning

Section 2.6 of Chapter 1 of the Passive ALWR Utility Requirements Document (URD) requires that passive ALWRs be designed to allow simplification and standardization of emergency planning. The plant designer is to perform an evaluation of the plant design against two ALWR emergency planning technical criteria as prescribed for containment performance and site boundary dose. The plant designer is also to perform a supplemental PRA evaluation in support of the evaluation against the two ALWR emergency planning criteria. The methodology which is specified in the URD for demonstrating the criteria is to be utilized in this evaluation. For ALWR emergency planning, the plant is to be provided with the capability to address severe accident containment challenges, including design features and characteristics to preclude core damage sequences which could bypass containment, and to withstand loads representing those associated with core damage sequences. ASME limits for structural integrity specified in Chapter 5, Section 6.6.2.2 of the URD should not be exceeded for a period of approximately 24 hours after the start of release of fission products from the fuel. Beyond approximately 24 hours, means for preventing uncontrolled fission product release from containment are to be provided in accordance with Chapter 5, Section 6.6.2.5 of the URD. Dose at the site boundary is not to exceed 1 rem for a period of approximately 24 hours from the start of release of fission products from the fuel. A PRA evaluation is to be performed to demonstrate that the following goals are met:

- A core damage frequency $\leq 10^{-5}$/yr;
- A cumulative frequency $< 10^{-6}$/yr for sequences resulting in greater than 1 rem over 24 hours at the site boundary.

In addition, it is to be demonstrated that ALWR designs are consistent with the prompt accident quantitative health objectives of the NRC safety goal policy with no credit for evacuation prior to 24 hours. EPRI intends that these criteria and methodology be used in an integrated manner, and the criteria are not to be applied without using the methodology as specified by EPRI in the URD.

The staff position regarding emergency planning for passive ALWR designs is discussed in Annex D of Appendix B to Chapter 1. The staff concludes that EPRI's requirements for emergency planning for passive ALWR designs, as described above, do not conflict with current regulatory guidance and are, therefore, acceptable. However, because the determination of the adequacy of an applicant's emergency plan is design and site-specific, it will be evaluated during the staff's review of an individual application for a combined licensed. Also, as discussed under Item III.G of Annex A of Appendix B to Chapter 1 of this report, the staff will evaluate any proposed reduction of offsite emergency planning requirements during its review of individual applications for FDA/DC.

# 3 PERFORMANCE DESIGN REQUIREMENTS

## 3.1 Introduction

Section 3 of Chapter 1 of the passive 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 passive Requirements Document applies to designs extending up to 600 MWe per unit.

## 3.3 Plant Design Life

Section 3.3 of Chapter 1 of the passive 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. Although the Commission paper was directed to the evolutionary designs, the staff concludes that it is equally applicable to the passive designs. It is the applicants' responsibility to identify the components and systems that are affected. The staff will address plant life during its review of individual applications for FDA/DC. These applications will have to provide information and programs to support design life and the staff's reviews of such issues as fatigue, corrosion, and thermal aging. The staff discusses this item further in Section 4.7.3 of this chapter.

## 3.4 Maneuvering and Response to Grid Demands

Section 3.4 of Chapter 1 provides EPRI-specified requirements for load following, frequency control, grid breakup, and load rejection.

## 3.5 Event Response Capability

Section 3.5 of Chapter 1 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 to hot standby at full pressure and temperature in 24 hours. Similarly, EPRI requires that the plant 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 originally required 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 will be determined based on the results of a plant-specific evaluation to address the concerns delineated in GL 86-09. In the DSER for Chapter 1, the staff stated that it would evaluate that analysis during its review of an individual application for FDA/DC. In Revision 4 of the passive Requirements Document, Section 3.5.2 of Chapter 1 regarding the requirement for secured coolant pump operation was deleted. The staff concludes that deletion of this requirement is acceptable and resolves the staff's concern in this area.

Section 3.5.3 of Chapter 1 requires that the plant response to reactor trips that 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 safety or relief valve, or the backup feedwater system 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 inadvertent closure of the main steam isolation valves (MSIVs) while at full power not result in the actuation of the safety/relief valves for BWRs. Section 3.5.6 requires that the loss of a running main feedwater or condensate pump while at full power not result in a reactor trip. Section 3.5.7 requires the systems and equipment to be designed to withstand a complete loss of bulk ac power for at least 2 hours without exceeding equipment design limits. 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.

## 3.6 Core Performance

Section 3.6 of Chapter 1 requires ALWR core designs that have a capability of up to a 24-month fuel cycle and fuel mechanical designs that have a peak bundle burnup of at least 50,000 and 60,000 megawatt-days per metric tons of uranium (MWD/MTU) for BWRs and PWRs, respectively. These valves are greater than NRC-approved fuel burnup levels at current operating plants. To support this high fuel burnup operation, each application for FDA/DC 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 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 application for FDA/DC 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 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.

## 3.8 Occupational Radiation Exposure

In 1991, the average annual occupational exposure dose for U.S. nuclear power plants was approximately 260 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 passive 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 materials selection 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 RG 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 passive Requirements Document includes the general structural design criteria and requirements applicable to all buildings, structures, systems, and equipment. It also addresses the structural requirements for passive features 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 passive 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 passive 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. 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 passive Requirements Document proposes the use of ANSI/ANS 51.1, "Nuclear Safety Criteria for the Design of Stationary PWRs," and ANSI/ANS 52.1, "Nuclear Safety Criteria for the Design of Stationary BWRs," as an alternative way of complying with RG 1.26. The staff has not completely endorsed these two industry standards for the evolutionary ALWR and concludes that, because of the unique designs of the passive ALWRs, neither of these standards will be applicable to these plants. As discussed in Section 4.31 of Appendix B to Chapter 1 of the passive Requirements Document, SRP Section 3.2.2, "System Quality Group Classification," also may not be completely applicable. Many of the passive systems that EPRI has classified as non-safety related are similar to systems in current LWR designs that are classified as safety related. However, these passive plant non-safety-related systems are relied on to provide defense-in-depth capabilities to serve as the first line of defense in the event of transients and plant upsets to reduce challenges to the passive safety systems. In its letter dated May 26, 1993, EPRI proposed an approach to determine which systems should receive which kind of regulatory oversight. A key element of this process is to ensure that the

passive plant vendors use probabilistic risk assessment methods to examine reliability of the safety and non-safety systems and ensure that important risk insights are considered in the design. This process, and the staff's evaluation of it, are described in greater detail in Section 2.2 of Chapter 3 and Section A of SECY-94-084 in Annex B of Appendix B to Chapter 1 of this report. Because the level of detail in the passive Requirements Document does not permit the staff to develop final positions on the safety and quality classifications of structures, systems, and components, the staff will review implementation of EPRI's process during its review of an individual application for FDA/DC to ensure that the passive plant vendors follow this process satisfactorily.

4.3.2 Seismic Classification

Section 4.3.2 of Chapter 1 of the passive 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 steam isolation valve 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 to this chapter.

Section 4.3.2 of Chapter 1 of the passive Requirements Document defines the seismic Category I items as including all structures, systems, and components whose safety class is SC-1, SC-2, or 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 passive 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.

Table 1.1 summarizes the staff's understanding of the ALWR seismic classifications and their relationship to safety classifications.

Table 1.1 ALWR Seismic Classifications and Their Relationships to Safety Classification

| 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 passive Requirements Document states that seismic classification will be consistent with RG 1.29. In NRC Review Guideline 17 in NUREG-0416, "Security Plan Evaluation Report," 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 staff guidelines contained in Section 13.6 of the SRP, RG 1.29 and NUREG-0416 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 (SSE) and to maintain the specified design functions.

Seismic Category II

In Section 4.3.2.2 of Chapter 1, EPRI defines a seismic Category II classification that is applicable 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. Positions C.2 and C.4 of RG 1.29 provide guidelines for the design and quality assurance of such items. Position C.2 states that these items should be designed and constructed so that the safe shutdown earthquake would not cause the type of failure or interaction described above. Position C.4 states that the pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 should be applied to these items. In Section 4.3.2 of Chapter 1 of the passive Requirements Document, EPRI provides a specific requirement that Positions C.2 and C.4 of RG 1.29 be applied to seismic Category II items. The broad commitment to RG 1.29, supplemented by the specific commitment to Positions C.2 and C.4, is acceptable. 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 to 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 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 current staff positions contained in RG 1.29 and SRP Sections 37.3 and 3.9.2 and is, therefore, acceptable.

Section 4.3.2.2.2 of Chapter 1 of the passive Requirements Document states that the plant designer will select appropriate approved 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. The rationale portion of this section includes the statement: "The seismic ductility factors and ductility limits are anticipated to consider the results of research which is documented by the time of the ALWR, and to provide significant improvements over current practice." The staff concludes that this requirement is acceptable because it ensures that plant designers will use methodology reviewed and approved by the staff during this part of the seismic design process.

Non-Seismic

Section 4.3.2.3 of Chapter 1 of the passive Requirements Document states that non-seismic (NS) structures are those that do not fall into seismic Category I or II definitions in Sections 4.3.2.1 and 4.3.2.2. Section 4.3.2.3 requires that NS structures and equipment be designed to the seismic requirements of Zone 2A of the Uniform Building Code (UBC). The building structures will be classified as "essential buildings with an importance factor of 1.25." Because many regions in the United States are designated as UBC seismic Zone 2B or higher, the staff stated in the DSER for Chapter 1 that the use of the Zone 2A specification in these zones was not acceptable for the design of any NS items. The staff stated that it would require the use of Zone 2B of the UBC in an application for FDA/DC. In it letter dated September 29, 1992, EPRI stated that it has defined a "Category II" as well as "Non-Seismic" categories. The non-seismic classification may only be applied to plant SSCs that perform no nuclear safety function and whose structural failure or interaction will not degrade the function of a seismic Category I structure,

system, or component and will not result in injury to occupants of the control room. On the basis of this understanding of EPRI's requirements, the staff concludes that the use of the UBC Zone 2A requirements with an importance factor of 1.25 for the design of these structures is acceptable.

## 4.4 Codes and Standards

Section 4.4 of Chapter 1 of the passive Requirements Document provides EPRI-proposed requirements relative to the applicability of major design and construction codes, industry standards, and regulatory positions to the ALWR passive plant design. Tables 1.4-2 and 1.4-3 in Chapter 1 list industry technical standards and major structural design and construction codes, respectively, that are applicable to the ALWR. Several of these standards have not been endorsed by the staff and should not be used as the basis for plant design and construction. In its letter dated August 1, 1991, EPRI agreed to revise 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 was to ensure that the staff's reviews of applications for FDA/DC for passive plants will be conducted using acceptance criteria that include those codes and standards most recently approved by the NRC. In Revision 2 to Section 4.4 of Chapter 1, this requirement was revised in accordance with the staff's request. The staff concludes that this commitment is acceptable. The staff will evaluate this matter during its review of an individual application for FDA/DC or COL.

In Revision 0 to the rationale portion of Section 4.4 of Chapter 1 of the Requirements Document for evolutionary plant designs, EPRI stated that it anticipated that the applicable edition of codes and standards would be that in effect approximately 42 months before start of construction. In the DSER dated September 1987 on Chapter 1 of the Requirements Document for evolutionary plant designs, the staff expressed concern 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 Subsection 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 plant construction permit is docketed.

In response to this concern, EPRI deleted the "42-month" criterion from the rationale portion in the revised Requirements Document. In addition, EPRI added a note to Table 1.4-1 in Chapter 1 of the Requirements Document 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.

See the regulatory departure analysis in SECY-93-087, included in Appendix B to this chapter, for additional staff discussion on the use of industry codes and standards.

4.4.1 Major Design and Construction Codes

In its letter dated May 17, 1991, the staff requested EPRI to revise Item g under "ASME Code" in Table 1.4-3 of Chapter 1 of the passive Requirements Document 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 passive 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.

4.4.2 Industry Technical Standards

Section 4.4.2 of Chapter 1 references Table 1.4-2, which lists the industry technical standards that will be applicable to the ALWR passive plant. The staff's comments relative to this table are contained in Section 4.4 above.

4.4.3 Regulatory Positions

In Section 4.4.3 and Appendix B of Chapter 1, EPRI identifies NRC RGs and SRP sections that will be applicable to the structural design bases of the ALWR passive plants. EPRI lists the applicable guidance in Table B.1-2 of Appendix B to Chapter 1 of the passive Requirements Document (EPRI's original reference to "Table 1" in Section 4.4.3 appeared to be a typographical error). 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.

Implementation of Leak-Before-Break Criteria

The staff discusses this issue in Section 4.5.5 and Appendix B of this chapter.

Damping Values

Section 4.4.3.3.3 of Chapter 1 of the passive 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 agrees that Code Case N-411 has been provisionally accepted for piping only. However, for design of buildings, structures, and equipment, the only staff-approved damping values are in RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants." In Appendix B to Chapter 1, EPRI requires that plant designers meet the regulatory guidance in RG 1.61. Therefore, the staff concludes that EPRI's request to use approved realistic damping values in the analyses of buildings, structures, and equipment is acceptable. This issue is also discussed in Section 4.7.3 of this chapter.

Elimination of Operating-Basis Earthquake From Design

In Section 4.4.3 of Chapter 1 of the DSER, the staff reported that, in Revision 1 to the passive Requirements Document, EPRI had deleted the operating-basis earthquake (OBE) for consideration in the design process. The staff further reported that, although it agreed in principle with EPRI regarding the deletion of the OBE from plant design, certain issues that are a direct consequence of eliminating the OBE need to be adequately resolved. This issue was identified as a part of Policy Issue 1.M, "Elimination of OBE." On April 2, 1993, the staff issued SECY-93-087, which included the staff's recommendations on this policy issue. The staff's evaluation of this issue is given in Section 2.1.1 of Appendix B to Chapter 1 of this report.

Use of the Independent Support Motion Response Spectrum Analysis Method

Section 4.4.3.3.4 of Chapter 1 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, "Dynamic Testing and Analysis of Systems, Components, and Equipment." 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. In the DSER for Chapter 1, the staff concluded that Sections 4.4.3.3.4 and 4.7.3.4 of Chapter 1 should be revised to provide this commitment. In Revision 4 and its letter dated May 3, 1993, EPRI responded to this issue by revising Section 4.7.3.4 to define a support group, consistent with the staff's position as stated above. The staff concludes that this is acceptable.

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. 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 passive 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.

Analysis of Vibratory Loads With Significant High-Frequency Input

In Sections 4.4.3.3.6 and 4.7.3.3 of Chapter 1 and Section 2.1.1 of Appendix B to Chapter 1, EPRI states that in the analysis of vibratory loads (other than seismic) with significant high-frequency input (i.e., 33 to 100 Hz), the plant designer may combine high-frequency modal results by algebraic combination. This is a deviation from RG 1.92, "Combining Modal Responses and Special Components in Seismic Response Analysis," that the staff currently evaluates on a case-by-case basis. In Revision 2 of the passive 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 in 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.

In its letter dated May 17, 1991, the staff requested EPRI to 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 to this request, 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

The staff discusses this issue in Section 4.8.1 of this chapter.

Design-Basis Tornado

Section 4.4.3.3.8 of Chapter 1 of the passive Requirements Document specific that Table 1.2-6 of Chapter 1 contains EPRI's. This issue is discussed by the staff in Section 4.5.2 of the chapter and in Section 2.1.2 and as Item II.F of Annex A to Appendix B of this chapter.

## 4.5 Design Loads and Conditions

### 4.5.1 Introduction

Section 4.5.1 of Chapter 1 of the passive 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, it is not currently 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 of the passive Requirements Document, including wind loadings, hydrology, geology and foundation conditions, earthquakes, tornadoes, and volcanic activity. This table provides the envelope of plant design parameters associated with these natural phenomena.

Wind Loadings

Table 1.2-6 of Chapter 1 specifies a basic wind speed of 110 mph based on a 50-year recurrence interval. This basic wind speed is to be scaled by an importance factor (as defined in ANSI A58.1-1982) 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 application 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 based on a 50-year recurrence interval is not acceptable to the staff because of a concern regarding the potential adverse effect of the collapse of non-safety-related structures (e.g., turbine building or stack). If the collapse of non-safety-related structures due to wind loading does not adversely impact the function of the safety-related structures, systems, and components, the use of 1.0 as the importance factor is suitable; if not, the importance factor of 1.11 should be used in the design of such structures. Therefore, the staff will evaluate the effects of wind-induced failure of non-safety-related structures on safety-related SSCs during its review of an individual application for FDA/DC.

Hydrology

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

- Maximum ground water level - The passive Requirements Document requires the maximum ground water level to be 2 feet below grade. In the DSER for Chapter 1, the staff stated that this requirement was not acceptable and that the maximum ground water level should be at grade. The staff did not feel that EPRI had provided adequate justification for its position. Therefore, the staff would address this item during its review of an individual application for FDA/DC. In its letter dated September 29, 1992, EPRI provided a clarification of its requirements for maximum ground water level, stating that the maximum flood level is set at 1-foot below the plant grade level, and a 1 foot allowance for wave runup is provided. Further, the plant grade level should be above the high ground water level to permit functioning of the storm drainage system. EPRI noted that this might necessitate a siting interface criterion so that a check is made that the standardized design assumptions are compatible with the selected site. The staff concludes that EPRI's clarification resolved its concern regarding maximum ground water level and that the related requirements of Table 1.2-6 are acceptable.

- Maximum flood (tsunami) level - The passive Requirements Document originally proposed that the maximum flood level be in the range of 1-foot below to 26 feet above grade. Although in the DSER dated September 1987 on Chapter 1 of the Requirements Document for evolutionary plants, the staff concluded that this design parameter was consistent with regulatory requirements, it recommended that the 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) - The passive Requirements Document originally specified a maximum rainfall rate of 10 in./hr and a maximum snow load of 50 lb per square foot (psf). In the DSER dated September 1987 on Chapter 1 of the Requirements Document for evolutionary plants, the staff raised a concern that the rainfall rate of 10 in./hr was much too low, stating that the probable maximum precipitation (PMP) in a 5-minute interval over 1 square mile is 6.3 inches in the Great Lakes area and 6.2 inches along the Gulf Coast. The revised Table 1.2-6 specifies a higher 1-square mile, 1-hour PMP of 19.4 inches, together with a 1-square-mile, 5-minute PMP of 6.2 inches. The 5-minute PMP value appears reasonable; however, that might exclude a number of Great Lakes area sites. EPRI should use the SRP guidelines and relevant RGs for developing an adequate structural and flood-prevention design basis for the PMP. 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.

  In the DSER dated September 1987 on Chapter 1 of the Requirements Document for evolutionary plants, 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, in Table 1.2-6 of Chapter 1 of the passive Requirements Document, the 50-psf snow load was retained. The staff concludes that this snow load is

unacceptable. 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.

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 its letter dated April 24, 1991, the staff requested EPRI to 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 bearing 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 passive Requirements Document, and that the ALWR objectives will be satisfied as long as the standard plant design is 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 issue during its review of an application for a COL.

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 Requirements Document, and 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 included in the Requirements Document, the staff concludes that the guidelines for minimum margin against liquefaction potential may be addressed by the applicant for a COL 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 the 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 EPRI to 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 COL.

Earthquakes

In the DSER, the staff reported that, in the process of eliminating the OBE for consideration in the design process, Sections 4.5.2.4.4 and 4.8.1.1 in Chapter 1 of the passive Requirements Document contain requirements for evaluating of fatigue and seismic qualification of equipment that reduce the number of full stress cycles for one-half SSE from the staff position of 50 to 20. The staff's evaluation of this issue identified this as part of Policy Issue I.M, "Elimination of OBE." On April 2, 1993, the staff issued SECY-93-087, which included the staff's recommendations on this policy issue. Included in these recommendations is the staff's position relative to the number of earthquake cycles to be used by the ALWR plant designer for the evaluation of fatigue and seismic qualification of equipment. These recommendations do not agree with the EPRI position. Therefore, the requirements in Sections 4.5.2.4.4 and 4.8.1.1 are not completely acceptable, and the staff will review individual applications for FDA/DC in accordance with SECY-93-087. The staff's evaluation of Policy Issue I.M is discussed in Section 2.1.1 of Appendix B to Chapter 1 of this report.

In Table 1.2-6 of Chapter 1, EPRI states that the ALWR plant should be designed for 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"
- time history to envelop the design response spectra

Regarding the SSE, Note 10 to 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 motion be deconvoluted to specific points for application 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 EPRI to provide a justification for 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 there are sufficient data indicating 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 a rock outcrop or a hypothetical rock outcrop in the free field. For the purpose of 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 state-of-the-art practice and the guidance of the NRC's RGs 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 a RG 1.60 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 recordings of earthquakes in the Eastern United States possess more high-frequency (greater than 5 Hz) ground motion than those earthquake records used to develop the RG 1.60 response spectrum. This could limit the sites at which designs using the 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. As specified in 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 to be used for the seismic analysis comply fully with the SRP. The staff will address this issue during its review of an individual application for FDA/DC.

Tornado

In Section 4.5.2.5 and Table 1.2-6 of Chapter 1 and Section 2.1.2 of Appendix B to Chapter 1, EPRI originally proposed that the maximum tornado wind speed of 260 mph and the tornado recurrence interval of 1 million years (1.0E-6 per year tornado strike probability) be used for the design-basis tornado. These parameters are based on ANSI/ANS 2.3, "Standard for Estimating Tornado and Extreme Wind Characteristics at Nuclear Power Sites." Section 2.1.2 of Appendix B to Chapter 1 lists tornado design for the passive ALWR as a plant optimization subject, which the staff discusses in the regulatory departure analysis of SECY-93-087 in Appendix B to 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. WASH-1300 states that the probability of occurrence of a tornado that exceeds the DBT should be on the order of 1.0E-7 per year per nuclear power plant and the RG delineates 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 RG was developed. The contractor completed its reevaluation, which 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 (NSSFC) with 30 years of data, 1954 through 1983. This data tape contains the data for the approximately 30,000 tornadoes that occurred during the 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 1.0E-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 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 passive Requirements Document. In the DSER for Chapter 1, the staff stated that EPRI should revise Table 1.2-6 accordingly and that it would confirm this change. In Revision 4 of the passive Requirements Document, EPRI revised Table 1.2-6 of Chapter 1 to show maximum tornado wind speed of 300 mph and deleted the reference to the tornado recurrence interval from the Requirements Document. Therefore, this confirmatory issue is closed.

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 RGs 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. Hence, the staff's acceptance 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.

In SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactors Designs", dated April 2, 1993, the staff stated that it will accept the tornado design basis of 300 mph proposed by EPRI. Table 1.2 shows the design-basis tornado parameters that are considered acceptable to the staff. Therefore, this open issue is closed.

Table 1.2 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 |

Volcanic Activity

In its letter dated April 24, 1991, the staff requested confirmation from EPRI that the ALWR site will not be selected in areas subjected to the effects of volcanic activity near the site. In its July 2, 1991, response, EPRI confirmed that the standard plants would not be sited in such areas and that site-specific design assessments will be necessary if a plant owner decides to locate a standard plant in such areas. 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. In Table 1.2-6 of Chapter 1, the ambient temperature was originally 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 would be used when they are derived from a probabilistic method and are associated with certain probabilities of exceedance.

Table 1.2-6 also provides a criterion requiring the site be such as to permit atmospheric heat rejection of cooling water system heat loads or to provide cooling water at the flow rates and temperatures to be 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," has 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, additional measures may be needed to provide protection against surface vehicle bombs. The Commission has approved amendment of Part 73 to address protection against a land vehicle bomb. The staff will address this issue during its review of an individual application for FDA/DC.

4.5.4 Plant Operating Loads

Section 4.5.4 of Chapter 1 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 valve opening and closing, 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.

Extreme plant condition loads include pressures, temperatures, forces, and vibrations due to transients and accidents not encountered in normal operation, but which 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 the broad requirements in Section 4.5.4 of Chapter 1 relative to plant operating loads are consistent with current applicable staff guidelines in SRP Section 3.9.3 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 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.

Leak Before Break

Section 4.5.5 of Chapter 1 specifies that to the extent practicable, fracture mechanics and mechanistic analysis through the application of leak-before-break (LBB) criteria will be used to minimize the need to evaluate the consequential pipe rupture effects.

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 of SECY-93-087 in Appendix B to this chapter provides the background and staff position on this issue. In this Commission paper, the staff concluded 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 verified during the COL phase by performing the appropriate inspections, tests, and analyses criteria (ITAAC) discussed herein.

In the DSER for Chapter 1, the staff stated that it had evaluated the passive Requirements Document for LBB application to ALWRs and concluded that it was not completely acceptable. Because the limitations and acceptance criteria for LBB applications in ALWRs were the same as those established for currently operating nuclear power plants, this was identified as an open issue. In

Revision 3, EPRI responded to this issue and deleted Section 4.5.5.3.2, which permitted such use of LBB in determining BWR suppression pool loads. The staff concludes that this is acceptable; however, consistent with the staff position on this issue, appropriate bounding limits must be established during the FDA/DC phase by the vendor and preliminary LBB analysis may be performed during the FDA/DC phase. These bounding values and preliminary analysis shall be verified when the as-built and as-procured information becomes available during plant construction. Specific details will need to be developed as the process is implemented during the first trial application by the vendor. This issue will be resolved during the review of each COL application; therefore, this open issue is closed.

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 for internal-missile generation. The staff has not endorsed ANSI/ANS 58.1. Therefore, it concludes that where differences exist between the above SRP sections and ANSI/ANS 58.1, the guidance of 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 Table B.1-2 of Appendix B to Chapter 1, EPRI commits to comply with SRP Section 3.4.1, "Flood Protection." Section 4.5.5.5 of Chapter 1 of the passive Requirements Document states that the plant designer will identify those systems and equipment that must be capable of normal operation under flooded conditions and will specify the design conditions for systems and equipment required to operate under 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 the DSER for Chapter 1, the staff concluded that the above commitment and the requirements in Section 4.5.5.5 provided the necessary requirements to protect structures and systems from the effects of internal flooding and were, therefore, acceptable. The staff stated that it would confirm that EPRI had added information about internal flooding to Table 1.2-4 of Chapter 1 of the passive Requirements Document as committed to in a letter dated May 22, 1991. In Revision 4, EPRI incorporated the information about internal flooding into Table 1.2-4 of Chapter 1. Therefore, this confirmatory issue is closed.

## 4.6 Load Combinations

### 4.6.1 Buildings and Structures

Section 4.6.1 of Chapter 1 of the passive 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 shown 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, Section CC, for the design of concrete containments but are not referenced in the passive Requirements Document, the staff requested EPRI to confirm its position regarding compliance. In its letter dated July 2, 1991, EPRI indicated that while it is impractical to list all applicable RGs and SRP sections in individual paragraphs of the passive Requirements Document, the requirement of compliance for the ALWRs is shown in Table B.1-1 of Appendix B to Chapter 1 of the passive Requirements Document. To ensure that the plant designer will use proper additional regulatory guidance for the load combinations, the staff's position is that 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 Institute (ANSI/ACI) 349 and ANSI/American Institute of Steel Construction (AISC) N690, 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 concrete Category I buildings and that the NRC has not approved the use of ANSI/AISC N690 for steel Category I structures. In its response of July 2, 1992, EPRI indicated that while it is impractical to list all applicable RGs and SRP sections in individual paragraphs of the passive Requirements Document, the requirement of compliance for the ALWRs is shown in Table B.1-1 of Appendix B to Chapter 1 of the passive Requirements Document. 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 Category I steel structures. The staff reviewed the acceptability of ANSI/AISC N690 and finds that the use of ANSI/AISC N690-1984 for the design, fabrication, and erection of safety-related structures in ALWR passive plants is acceptable when supplemented by the following provisions:

(1) In Section Q1.0.2, the definition of secondary stress should apply to stresses developed by temperature loading only.

(2) The following notes should be added to Section Q1.3:

> When any load reduces the effects of other loads, the corresponding coefficient for that load should be taken as 0.9, if it can be demonstrated that the load is always present or occurs simultaneously with other loads. Otherwise, the coefficient for that load should be taken as zero.
>
> Where the structural effects of differential settlement are present, they should be included with the dead load "D."
>
> For structures or structural components subjected to hydrodynamic loads resulting from LOCA and/or safety/relief actuation, the consideration of such loads should be as indicated in the appendix to SRP Section 3.8.1. Any fluid structure interaction associated with these hydrodynamic loads and those from the postulated earthquake(s) should be taken into account.

(3) The stress limit coefficients (SLCs) for compression in Table Q1.5.7.1 should be as follows:

- 1.3 instead of 1.5 [stated in footnote (c)] in load combinations 2, 5, and 6.
- 1.4 instead of 1.6 in load combinations 7, 8, and 9.
- 1.6 instead of 1.7 in load combination 11.

(4) The following note should be added to Section Q1.5.8:

> For constrained (rotation and/or displacement) members supporting safety-related structures, systems, or components, the stresses under load combinations 9, 10, and 11 should be limited to those allowed in Table Q1.5.7.1 as modified by Provision 3 above. Ductility factors of Table Q1.5.8.1 (or Provision 5 below) should not be used in these cases.

(5) For ductility factors $\mu$ in Sections Q1.5.7.2 and Q1.5.8, substitute provisions of Appendix A to SRP Sections 3.5.3.II.2 in lieu of Table Q1.5.8.1.

(6) In load combination 9 of Section Q2.1, the load factor applied to load $P_a$ should be $1.5/1.1 \approx 1.37$, instead of 1.25.

(7) Sections Q1.24 and Q1.25.10 should be supplemented with the following requirements regarding painting of structural steel:

- Shop painting to be in accordance with Section M3 of Reference 1.

- All exposed areas after installation to be field painted (or coated) in accordance with the applicable portion of Section M3 of Reference 1.

- The quality assurance requirements for painting (or coating) of structural steel to be in accordance with Reference 2 as endorsed by Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water Cooled Nuclear Power Plants."

Decoupling of SSE and Pipe Rupture Loads for Buildings and Structures

Section 4.6.1.3 of Chapter 1 specifies that when the leak-before-break (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 if the effects (nondynamic) of a double-ended guillotine break (DEGB) and of an SSE are combined 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 passive Requirements Document excludes the OBE load terms from load combinations. The staff generally agrees with the approach to decouple the OBE from the SSE and 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, Section CC, require that SSE and loss-of-coolant-accident (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 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 technique. 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 DEGB of reactor coolant loop piping was postulated for containment sizing and emergency core cooling system 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).

In 1986, the NRC staff proposed a revision to SRP Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures," that would have allowed the decoupling of seismic and PWR primary loop pipe rupture loads for ASME Code Class 1, 2, and 3 components and their supports. Emphasis was given to limiting this revision to the mechanical design of components and their supports only for PWR primary coolant loop piping.

In parallel, the NRC was in the process of a rule change to General Design Criterion 4 of 10 CFR Part 50, Appendix A, to allow the application of the LBB method to piping systems. The planned revision would eliminate the need to design for dynamic effects of postulated pipe breaks, including pipe whip restraints and jet impingement shields and design of 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, then the combination of SSE loads and LOCA loads effectively becomes the SSE loads alone.

The NRC Piping Review Committee stated in Volume 5 of NUREG-1061 that the LBB approach was acceptable and recommended decoupling SSE loads and loads from PWR primary coolant loop pipe ruptures for the mechanical design of all ASME Class 1, 2, and 3 components and supports. However, the staff found that if SRP Section 3.9.3 were to be revised without also revising SRP Sections 3.8.1 through 3.8.4, an inconsistency would be introduced in the SRP that would have to be based on rulemaking. SRP Sections 3.8.1 through 3.8.4 deal with load combinations for structures (including the containment), whereas SRP Section 3.9.3 deals with load combinations for ASME Code Class 1, 2, and 3 components, supports, and core support structures. Thus, the decoupling of seismic loads and pipe rupture loads for only mechanical components was found to be inappropriate because of the rulemaking conflict.

Another possibility was to expand the scope of the SRP revision to structures as well as mechanical equipment. However, the staff found that there was an insufficient technical basis to expand 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. While there exists an adequate basis 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 that 1.0E-6 per reactor-year (for other than LOCA plus SSE) is acceptable to the staff in principle; however, the staff will only approve such an elimination 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. In its letter dated September 29, 1992, EPRI responded to this issue by stating that the industry position is that seismic Category I structures (other than the containment) and their subcompartments need not be designed for the combined effects of SSE and LOCA if the probability of occurrence is less than 1.0E-6 per reactor-year. This position is not consistent with the above staff position, which states that this probabilistic approach is not applicable to the LOCA plus SSE loading combination. Therefore, EPRI's response is not acceptable.

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, the proposal to decouple LOCA and SSE loads for equipment and systems is not acceptable at this time because of the inconsistency that would be created in the SRP and 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 Effect of Other Loads

Section 4.6.1.4 of Chapter 1 requires that consideration be given to the treatment of loads that reduce the effect 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 has not justified the use of the coefficient 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 justification for this position, 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 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 flotation. The passive Requirements Document references the minimum factors of safety that are specified in SRP Section 3.8.5, "Foundations." However, in its letter of April 24, 1991, the staff stated that 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 this commitment was properly implemented in Revision 2 to the passive 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 discussed by the staff in Section 4.3.2.3 of this chapter. However, the staff concludes that, as far as load combination is concerned, the use of the UBC for non-seismic structures is acceptable, because the staff does not require applicants to analyze them.

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 EPRI to justify using NUREG/CR-3876 methodology in defining loads and load combinations without a corresponding definition of the reliability index and consistent acceptance criteria. The staff also requested 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 passive 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 section in Revision 2 of this section also was revised to acknowledge that the use of probabilistic load combinations is a departure from the ALWR 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 passive 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, such an elimination will be approved by the staff 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. In its letter dated September 29, 1992, EPRI responded to this

issue by stating that the industry position is that this probabilistic approach can be used to take advantage of load combinations that decouple SSE and LOCA for the design of seismic Category I structures (other than the containment) and all seismic Category I subcompartments. This position is not consistent with the above staff position, which states that this probabilistic approach is not applicable to the LOCA plus SSE loading combination. Therefore, EPRI's response is not acceptable. The staff will evaluate this issue during its review of an individual application for FDA/DC.

4.6.2 Systems and Equipment

Sections 4.6.2.3 and 4.6.2.5 and Tables 1.4-5 and 1.4-7 of Chapter 1 have eliminated the loading combination of SSE and LOCA on the basis of recommendations given in NUREG-1061, Volume 4. NUREG-1061 recommends the elimination of this loading combination only for piping systems in the majority of PWRs. The staff has not endorsed this recommendation. The current staff position, as stated in Appendix A to SRP Section 3.9.3, is that SSE and LOCA loads should be combined and included in the Service Level D loads. This position is applicable to both BWRs and PWRs. 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 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 they are 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. In its letter dated September 29, 1992, EPRI responded to this issue by quoting portions of the staff's DSER for the ABWR. These quotes are contained in a letter dated September 11, 1992, in which the staff submitted a DSER to General Electric Company relative to the use of a single earthquake design for the ABWR. In this letter, the staff stated (1) "Based upon recent dynamic tests conducted by the EPRI and NRC, it has been demonstrated that the piping can withstand seismic inertial loading higher that an SSE without rupturing," and (2) ". . . the staff concludes that no replacement earthquake loading should be used to establish postulated pipe break locations." On the basis in part, of these statements, EPRI concludes that it is apparent that the staff acknowledges the unlikely occurrence of a LOCA concurrent with an SSE at least for the design of piping systems and recommends that the staff either revise the GDC criteria or permit exemption from GDC 2 for the ALWR designs during the vendor FDA/DC review process.

With respect to (1) above, it should be pointed out that the staff's position on combining SSE and LOCA does not assume that the SSE initiates a LOCA in a seismic Category I pipe. Rather, they are considered as two independent events. The staff implemented this conservative position to satisfy GDC 2 and

to provide sufficient margin in the design of applicable items to account for unanticipated loads and/or degradation. With respect to (2), the earthquake loading referred to is the operating-basis earthquake, not the SSE, and the statement was written in the context of postulated pipe breaks to satisfy SRP Section 3.6.2 ("Determination of Rupture Locations and Dynamic Effects Associated With the Postulated Rupture of Piping"), and is not related to a Level D loading combination to be used for design. With respect to the EPRI recommendation, the staff has no current plans either to revise GDC 2 or to permit exemptions from that GDC for this loading combination. 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 of the vendor- and utility-specific item 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 passive 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 the passive Requirements Document. In its letter dated July 2, 1991, EPRI noted that specific examples were identified in Section 4.4.3 of 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, ACI 349 and ANSI/AISC N690 as supplemented by the AISC load and resistance factor design (LRFD) method. For abnormal or extreme load combinations, the passive 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 specified in SRP Section 3.8.1, "Concrete Containment," 3.8.2, "Steel Containment," or 3.8.3, "Concrete and Steel Internal Structures of Steel or Concrete Containments."

In its letter dated April 24, 1991, the staff commented that the NRC had 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 of this requirement was necessary. Revision 2 of the passive Requirements Document was revised to state 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 had 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 RGs and SRP sections, and that Table B.1-1 of Appendix B to Chapter 1 of the passive Requirements Document confirms EPRI's commitment to comply with the regulatory positions, except for those analysis techniques associated with the optimization subjects. In the DSER for Chapter 1, the staff stated that EPRI's response and the requirements in Section 4.7.2.3 of Chapter 1 were not acceptable because ASCE 4-86 had not been reviewed and approved by the staff. Therefore, the staff concluded that the SRP and RG guidelines should be used for the plant analysis and design of future ALWRs.

In its letter dated September 29, 1992, EPRI stated that the NRC should review ASCE 4-86 and define specific areas of disagreement. After considering EPRI's response, the staff concludes that plant designers proposing to use ASCE 4-86 should submit a request for the staff review and approval on a case-by-case basis. The staff will evaluate the adequacy of ASCE 4-86, if requested, during its review of an individual application for FDA/DC or COL.

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 passive Requirements Document did not indicate the need for NRC approval of methods such as the spectrum-to-spectrum generation procedure, the staff requested a complete discussion of the limitations and verifications for such procedures. In its response of July 2, 1991, EPRI did not address the NRC concern sufficiently. The staff position is that all analysis methods used for licensing 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 the basis for using less conservative damping values. In its response of July 2, 1991, EPRI cited cable trays and hangers as an example for which extensive tests have been done by some utilities to generate more realistic damping values than those provided in RG 1.61, "Damping Valves for Seismic Design of Nuclear Power Plants." The staff position of using structural damping for the design of structures is described in RG 1.61. However, the staff will not exclude the use of structural damping based on test results. Therefore, the staff 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 originally required 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 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. In the DSER for Chapter 1, the staff concluded that EPRI had not provided sufficient justification for this deviation; therefore, the SRP guidelines should be used for the design of masonry walls located in the Category I structures of the future ALWRs. In addition, the staff strongly discourages the use of masonry walls in Category I structures as load-carrying members or non-load-carrying members. In the DSER, the staff stated that it would evaluate this issue during its review of an individual application for FDA/DC. In its letter dated September 29, 1992, EPRI stated that it agreed with the staff position that masonry walls must not be used as load-carrying members of a seismic Category I structure. In Revision 4, EPRI revised Section 4.7.2.7 to specify that the masonry walls that are to be used as partitions (the locations where the building structural frame has been erected) and removable wall panels be

designed as reinforced masonry in accordance with the guidance in SRP Section 3.8.4 ("Other Seismic Category I Structures"), Appendix A, and all reinforced masonry walls be constructed and inspected to the applicable requirements of the UBC. The staff concludes that this is an acceptable requirement and this open issue is closed.

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 the expansion anchors are used, the NRC requires the use of the conservative safety factors of Inspection and Enforcement Bulletin (IEB) 79-02 to account for uncertainty in field installation. In its response of July 2, 1991, EPRI acknowledged that it intended that expansion bolts be used only when necessary and that the expansion bolts 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. In the DSER for Chapter 1, the staff concluded that this response was not acceptable because there is no assurance that the IEB 79-02 safety factors are not applicable to Maxibolts. In its September 23, 1992, submittal, EPRI provided its basis for demonstrating that the IEB 79-02 inspection methods and safety factors are applicable to wedge or sleeve-type anchors, which are permitted only for small loads. For applications requiring substantial pullout capacity, Maxibolts would be used only if embedded anchorage was not preengineered. The Maxibolt installation procedure includes a tensioning of each bolt installed that serves as a proof test of the pullout capacity. For this reason, EPRI concluded that additional testing of Maxibolts was not required. After reviewing EPRI's response, the staff finds that because the tension applied to the bolt during the installation is only the tensile force for setting up the bolt and is much lower than the bolt capacity, the tensioning applied to each bolt during installation cannot serve as a proof test of the pullout capacity.

The staff concludes at this time that a safety factor of 4 should be used for concrete expansion anchors including the Maxibolt design. It recognizes that a lower safety factor may be justified on the basis of qualification testing under actual field conditions, and, therefore, will allow plant designers to propose a lower safety factor thus qualified on a case-by-case basis. Therefore, plant designers should follow the guidelines of IEB 79-02 and use a safety factor of 4 for Maxibolts. As an alternative, plant designers should submit to the NRC 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 2 to the passive 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 Section NE, the staff requested that EPRI alert the plant designer to the application of Code Case N-284 to asymmetric containments with large openings or provide specific conditions under which N-284 can be used. In its response of July 2, 1991, EPRI stated that experienced professional containment vessel designers, working under careful review of the utility owners and regulators, will not apply Code Case N-284 where provisions of the code case do not apply. EPRI also stated that it was beyond the scope of the passive Requirements Document to explain the limitations of the specific code case. In the DSER for Chapter 1, the staff stated that this was not acceptable. It concluded that NE requirements should be used for the evaluation of shell-type structures. As for Code Case N-284, the staff stated that it would evaluate its applicability during its review of an individual application for FDA/DC. In its letter dated March 31, 1993, EPRI proposed to change the Chapter 1 requirement to apply ASME Code Case N-284 for the evaluation of local buckling of ellipsoidal heads with internal pressure and that any other proposed methods must be identified by the designer for NRC approval.

The staff's evaluation of the acceptability of ASME Code Case N-284 for local buckling due to internal pressure in the containment shell areas of ellipsoidal or torishperical heads is described in Section 6.6.2.2 of Chapter 5 of this report. Therefore, this issue is considered subsumed by the staff's evaluation of Section 6.6.2.2 of Chapter 5. However, any other proposed methods should be identified by the designer for NRC approval.

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 requirements of the UBC. 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 more detailed discussions of design methodology requirements for systems and equipment than those in Section 4.4 of Chapter 1. The staff's evaluations of several of these requirements are 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 this document. In its letter dated May 17, 1991, the staff requested EPRI to delete the reference to Appendix N and to reference applicable RGs, 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 in Section 4.4.3 of this chapter. The use of Code Case N-397 is only one of several issues that either 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. Therefore, the staff will evaluate the use of Appendix N to ASME Code, Section III, during its review of an individual application for FDA/DC in accordance with applicable SRP sections.

Use of 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 of Chapter 1 of the passive Requirements Document, this additional sentence was unnecessary. In the DSER for Chapter 1, the staff reported that, since it assumed that a requirement in any section of the passive Requirements Document could override a commitment in Appendix B to Chapter 1, EPRI's response was unacceptable. In its letter dated September 29, 1992, EPRI responded to this issue by stating that the industry position is that commitments in Appendix B cannot be overridden by other parts of the passive Requirements Document. The staff concludes that this is an acceptable response, and its review of the passive Requirements Document assumes that Chapter 1, Appendix B, contains bona fide commitments. Therefore, since Appendix B contains a commitment to comply with RG 1.84, Code Cases N-411 and N-420 will only be used in separate analyses and as further conditioned by RG 1.84.

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 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 Section 4.7.3.8 be revised 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 in the design process and that 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 2.1.1 of Appendix B to 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 the requirement portion of Section 4.7.3.8 be revised 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. In the DSER for Chapter 1, the staff reported that, since it assumed that a requirement in any section of the passive Requirements Document could override a commitment in Appendix B to Chapter 1, EPRI's response was unacceptable. In its letter dated September 29, 1992, EPRI responded to this issue by stating that the industry position is that commitments in Appendix B cannot be overridden by other parts of the passive Requirements Document. The staff concludes that this is an acceptable response, and its review of the passive Requirements Document assumes that Chapter 1, Appendix B, contains bona fide commitments. Therefore, since Appendix B contains a commitment to comply with RG 1.84, Code Case N-411 will only be used as conditioned by RG 1.84.

Seismic Evaluation and Design of Small-Bore Piping

Section 4.7.3.13 of Chapter 1 states that seismic Category 1 piping 2 inches and less in nominal diameter will be analyzed based on the reference spectrum approach in NCIG-14, EPRI NP-6623, "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. In the DSER for Chapter 1, the staff stated that it was reviewing this document as a topical report, which was submitted by the Nuclear Management and Resources Council (NUMARC) in its letter dated March 19, 1991. The staff stated that pending completion of this review, its position was that the methodology in EPRI NP-6623 was not acceptable. This was identified as an open issue. The staff completed the first phase of its review and submitted a request for additional information (RAI) in a letter to NUMARC dated June 1, 1992. Although NUMARC reported in a letter dated November 23, 1992, that it would submit its response to the staff's RAI in early 1993, this information has not yet been submitted. Until NUMARC submits a satisfactory response to the staff's RAI, the staff's position remains that the methodology in EPRI NP-6623 is not acceptable. The staff will review the design of small-bore piping against the same criteria as those for large-bore piping during its review of an individual application for FDA/DC. Therefore, this open issue is closed.

Use of ASME Code, Section III, Subsection NF, and ANSI/AISC 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 RAI on this issue, 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 originally stated that core support structures would be designed to the criteria specified 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. In the DSER for Chapter 1, the staff noted that the requirement still contained the words "designed to" rather than "constructed to," which was not completely acceptable. In the DSER, the staff stated that during its reviews of individual applications for FDA/DC, it would 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. In Revision 4, EPRI revised Section 4.7.3.23 of Chapter 1 to require that core support structures be constructed to the criteria specified in ASME Code, Section III, Subsection NG. The staff concludes that this is an acceptable requirement, with the understanding that "construction" is as defined in Subsection NG-1110.

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 passive 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 to this request, 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.

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 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 affect 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 standard safety analysis reports should include a proposed approach for accounting for environmental effects in the fatigue analyses for all ASME Class 1, 2, and 3 systems, components, and equipment that are designed for a 60-year lifetime. This was identified in the DSER for Chapter 1 as a vendor- or utility-specific item. In a letter dated September 23, 1992, EPRI responded to this issue by providing a brief discussion on the status of the ongoing Pressure Vessel and Research Council (PVRC) program to review methods for fatigue life prediction in light of the limited data mentioned above. EPRI stated that the preliminary assessment of the PVRC is that the ASME Code procedures are generally conservative and that immediate and drastic changes to the code rules for fatigue design are not warranted. EPRI further stated that there is broad agreement within the PVRC to pursue development of a nonmandatory appendix that will provide ASME-approved procedures for evaluating environmental effects on fatigue. This appendix would be developed within 1-2 years and could be used in conjunction with the design acceptance criteria for the detailed piping and component designs as part of the FDA/DC process for eventual use by the COL applicant.

The staff has been participating in the PVRC program, but does not share the same degree of optimism implicit in the EPRI response. On September 29, 1992, the chairman of the PVRC Steering Committee on Cyclic Life and Environmental Effects in Nuclear Applications presented a progress report to the Board of Nuclear Codes and Standards. Many members of the steering committee and the working groups supporting the steering committee disagreed with the characterizations in the progress report, which appeared to minimize or understate the environmental effects. There does not appear to be a consensus within the PVRC at this time regarding the environmental effects on fatigue and any associated changes in the ASME Code. With respect to the proposed nonmandatory appendix, it is the staff's understanding that this is only one of several possible avenues that the PVRC is considering for addressing this issue. Considering the apparent lack of agreement within the PVRC at this time, it is unlikely that such a document could be completed within 1-2 years.

In any case, the staff concludes that this issue is beyond the scope of the passive Requirements Document because the document does not provide details of a specific design along with the extensive test data necessary to resolve this issue. Therefore, the staff will address it during its review of an individual application for FDA/DC.

Conclusion

The staff concludes, with the exception of the vendor- or utility-specific items that will be evaluated during the staff's review of an individual application for FDA/DC, as noted above, that the requirements in Section 4.7.3 of Chapter 1 of the passive Requirements Document relative to design methodology for systems and equipment are consistent with applicable SRP sections, RGs, and staff positions and are acceptable.

## 4.8 Testing and Qualification

Section 4.8 of Chapter 1 of the passive 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

Section 4.8.1 of Chapter 1 contains broad and generally acceptable requirements relative to seismic and dynamic qualification of mechanical and electrical equipment 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 by using the seismic experience database. In addition, Revision 2 to 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. The staff's position relative to the use of the seismic experience database is 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 those systems and components 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 staff has not approved the use of the earthquake experience database methodology as a seismic qualification method. Rather, the NRC has concluded that the NRC-approved, experience-based methodology, when used in conjunction with appropriate restrictions and caveats as discussed below, may be an

acceptable means of verifying the seismic adequacy of certain equipment in certain operating nuclear power plants. The extent to which the staff 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 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 as discussed in the staff's resolution of USI A-46, 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.

In the DSER for Chapter 1, the staff reported that in Section 4.8.1.8 of Chapter 1 of the passive Requirements Document, EPRI formerly proposed to use the Seismic Qualification Utility Group's Generic Implementation Procedure (GIP) as one means of seismically qualifying equipment. The staff further reported that it did not accept the GIP as a qualification procedure for new plants because the GIP is considered by the staff to be a verification procedure and is intended to be used only by older operating plants under the USI A-46 resolution. The development of the GIP was not necessarily based on the required elements of IEEE 344-1987 or staff positions for newer operating plants.

In Revision 2 to the passive Requirements Document, EPRI revised Section 4.8.1.8 to delete the reference to the GIP and provide a requirement that the use of experience data for equipment qualification be in accordance with IEEE 344-1987. The rationale portion of Section 4.8.1.8 states that RG 1.100, Revision 2, indicates that use of the seismic experience database method in IEEE 344-1987 may require NRC evaluation for acceptance on a case-by-case basis. The staff concludes that the above requirement and rationale, together with the commitment to RG 1.100, Revision 2, in Chapter 1, Appendix B, Table B.1-2, provide reasonable assurance that the ALWR plant designer will not use the seismic experience database for equipment qualification before the staff reviews and approves the proposed methodology.

Therefore, consistent with RG 1.100, Revision 2, the staff will evaluate the use of experience data on a case-by-case basis during its review of an individual application for FDA/DC or COL.

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, which reduces the number of full-stress cycles for one-half 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

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.

Conclusion

The staff concludes that the requirements in Section 4.8.1, 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

EPRI has committed in Table B.1-2 of Appendix B to Chapter 1 to comply with SRP Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment." The staff finds this commitment acceptable. Section 4.8.2.1 of Chapter 1 of the passive Requirements Document states that Class 1E electrical equipment will be environmentally qualified in accordance with 10 CFR 50.49, "as outlined in IEEE 323." The staff finds that IEEE 323-1974 is acceptable as outlined in 10 CFR 50.49, but IEEE 323-1983 has not been found acceptable by the staff. 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. Therefore, in the DSER for Chapter 1, the staff stated that the above statement in Section 4.8.2.1 of the passive Requirements Document was not acceptable. The phrase "as outlined in IEEE 323" should be deleted from the sentence. This was identified as an open issue. In a letter dated September 29, 1992, EPRI restated its position that for all codes and standards, it will require designers to use the latest edition unless written justification is provided to use a different edition. In this case, EPRI is specifying that the latest edition be used, even though it is inconsistent with the current regulation. Plant designers and utilities, when using the passive Requirements Document, should be aware of the above differences between EPRI and NRC positions. The staff concludes that where differences exist between the regulations in 10 CFR 50.49 and the latest edition of IEEE 323, the requirements in the regulation should be satisfied as a minimum. If a plant designer identifies and provides justification for the

difference, 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. Therefore, this open issue is closed. The staff's position on the use of the latest codes and standards is presented in greater detail in Section II.A of SECY-93-087, which is contained in Annex A of Appendix B to Chapter 1 of this report.

EPRI states in Table B.1-2 of Appendix B to Chapter 1 that RG 1.89, "Qualification of Class 1E Equipment for Nuclear Power Plant," 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 are 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 must be qualified for its intended service and 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. In the DSER for Chapter 1, the staff found that the above statement could easily be misinterpreted, and therefore, it needed to be clarified by stating that the method of qualification should be in accordance with 10 CFR 50.49(f). Section 50.49(f) of 10 CFR requires each item of electric equipment important to safety be qualified by one of the following methods:

(1) 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

(2) testing a similar item of equipment with a supporting analysis to show that the equipment to be qualified is acceptable

(3) experience with identical or similar equipment under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable

(4) analysis in combination with partial type test data that supports the analytical assumptions and conclusions

This was identified as an open issue. In Revision 4, EPRI revised Section 4.8.2.4 to require that Class 1E electrical equipment be qualified following the methodology stated in 10 CFR 50.49(f). The staff concludes that this is acceptable; therefore, this open issue is closed.

# 5 MATERIALS

## 5.1 Introduction

Section 5 of Chapter 1 of the passive Requirements Document provides general guidance regarding material selection and the methods that will be used to fabricate nuclear power plant components. The requirements are intended to prevent materials degradation 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 passive Requirements Document requires that the materials for the ALWR be specified by the plant designer and that the materials used 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 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 RG 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 passive

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 where 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 for which 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 and are, therefore, acceptable.

5.2.6 Review of LWR Experience

Section 5.2.6 of Chapter 1 requires that the plant designer consider the lessons learned from the operation of existing plants and the measures necessary to prevent problems from recurring in the selection of ALWR materials. The ALWR plant designer will conduct a specific review of LWR experience (both domestic and foreign) to identify significant materials problems and identify proven methods of resolving them. This review will supplement and update the review of experience that is inherent in the passive Requirements Document. The review 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 material 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, such as austenitic stainless steel or carbon and low-alloy steels, be 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, 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 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 FDA/DC 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 from incomplete removal 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, in the DSER for Chapter 1, the staff concluded that EPRI should revise the passive 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 be included. This was identified as an open issue. The NRC staff has discussed this issue with EPRI. Those discussions established that only one U.S. nuclear steam supply vendor requires in its proprietary specifications that the total nitrites be kept below 10 ppm and total nitrates be kept below 820 ppm. No other technical basis was found to confirm that a limit on nitrites and nitrates might lead to prevention of corrosion in components during their service life. Therefore, the staff concludes that limits on nitrites and nitrates in the fabrication and construction of metal components are not warranted. Therefore, this open issue is closed.

## 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 passive 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.

- Cold-work control of material 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 (Rockwell Scale B hardness).

  - 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 supplied with the proper amount of cold work.

- EPRI originally stated that the following manual grinding controls were applicable to a BWR and were 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 needs, or to remove scratches or surface defects.

  - All grinding on 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 will be determined on undiluted weld deposits by magnetic measurements as specified in ASME Code, Section III, Division 1 (Winter 1976 or later).

In the DSER for Chapter 1, the staff concluded that these requirements were acceptable and their implementation would ensure that wrought austenitic stainless steel would perform in service as designed. The staff stated, however, that the grinding controls should also be applied to PWRs and that it would evaluate this issue during its review of an application for a COL. In

Revision 4, EPRI revised Section 5.3.1.1.5 to require that grinding controls be applied to PWRs as well as BWRs. The staff concludes that the requirements of Section 5.3.1.1 are acceptable.

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 for 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.

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

The staff considers these requirements acceptable. Their implementation 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 Alloy 600 because of its special properties (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 and 690 will be given a special thermal treatment to improve resistance to stress corrosion cracking. Alloy 690 must be used for steam generator tube applications.

For BWR applications requiring the use of Ni-Cr-Fe alloys because of their special properties (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 or 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.

In the DSER for Chapter 1, the staff concluded that these requirements were acceptable and their implementation would ensure that Ni-Cr-Fe alloys would perform in service as designed. However, the staff stated that it would require that the applicant for any standard design application identify the use of Alloy 600 and provide described information concerning its use. Those applications would 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. Those applications would be reviewed on a case-by-case basis. In its letter dated September 29, 1992, EPRI stated that for PWRs, the passive Requirements Document allows the use of Alloy 600 only outside the pressurizer and where low carbon stainless steel cannot be used. Written justification will be required of the designer. Alloy 690 will be restricted to steam generator tube applications.

EPRI further stated that for BWRs, Ni-Cr-Fe alloys are considered not completely proven material. The use of this material for specific applications (e.g., stub tubes in the reactor pressure vessel) will be allowed with written justification.

The staff concludes that EPRI's response is consistent with the staff's position. The staff will review individual applications for FDA/DC to ensure that the use of Alloy 600 and other Ni-Cr-Fe alloys such 690 or 800 is limited to acceptable locations.

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.

- All austenitic stainless steel castings that operate in water or steam environment at a temperature of 200 °F or greater in borated water or that are part of the RCS pressure boundary will have a ferrite content ranging from 8 percent to 30 percent. The ferrite content will be determined using American Society for Testing Materials (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 and this will be demonstrated by the 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 percent.

  - 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 that 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 considers these requirements acceptable. Their implementation 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. The designer must technically justify and document that unclad items are not susceptible to serious forms of corrosion under the full range of environmental operating conditions to which the item will be exposed, or adequate means must be provided, such as oxygen injection, to ensure that the unclad surfaces are protected against corrosion. If oxygen injection is used, the designer must document that integranular stress corrosion cracking (IGSCC) and irradiation-assisted stress corrosion cracking (IASCC) protection provided by hydrogen water chemistry for RPV and non-replaceable internal components is not compromised. The RPV must be clad. 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 requires 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 of 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 of 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.

- Conclusion

  The staff considers the requirements of Section 5.3.1.5 of Chapter 1 acceptable. Their implementation will ensure that carbon and low-alloy materials will 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 ASME 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 (SCC). 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 considers these requirements acceptable. Their implementation will ensure that precipitation-hardened stainless steel will perform in service as designed.

Ni-Cr-Fe Alloy X-750

Sections 5.3.1.3.5 and 5.3.1.3.1 of Chapter 1 specify 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 where a substantial base of successful experience exists. If 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 Alloy X-750.

Prevention of Intergranular Stress Corrosion Cracking (IGSCC) of Austenitic Stainless Steel

Section 5.3.1.8 of Chapter 1 specifies that austenitic stainless steel must be resistant to 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 requirements below.

- 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 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 5.4.3.4 of Chapter 1 of the passive Requirements Document.

- Surfaces of all components, and parts, and raw material, stock which have been heat treated or pickled, 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.

The staff considers these requirements acceptable. Their implementation will ensure that austenitic stainless steel will be resistant to IGSCC in service. The staff considers it important that adequate field and shop fabrication processes be used to minimize the sensitization of materials to IGSCC and will review specific controls for conformance with the recommendations of RGs 1.31, 1.34, 1.36, 1.37, and 1.44 as a part of the COL process.

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 are 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 "Guide to the Design of Secondary Systems and Their Components To Minimize Oxygen-Induced Corrosion" (EPRI NP-2294).

- 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 not be used for this application.

The staff considers these requirements acceptable. Their implementation 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 threaded fasteners used inside those systems and in pipe and component structural mountings for those systems will be selected and specified 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, 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 passive 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 maintain pressure boundary integrity in the reactor coolant and related systems and in the steam, feedwater, and condensate systems and 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 considers these requirements acceptable. Their implementation 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 makes non-metallic materials unsuitable.

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

The staff considers these requirements acceptable because metallic insulation has proven to perform satisfactorily in service. Non-metallic insulation will be used in accordance with the guidance in RG 1.36.

### 5.3.5 Seals, Gaskets, Packing, Sealants, Paints and Protective Coatings, Lubricants and Hydraulic Fluids, 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 nuclear steam supply systems (NSSSs) and other applications in the ALWR. The designer will rely on information from component vendors only when substantiated by operational experience.

The staff considers these requirements acceptable. Their implementation will ensure that seals, gaskets, packing, sealants, paints and protective coatings, lubricants and hydraulic fluids, cleaning, packaging, and storage materials are selected on the basis of operational experience. However, in the DSER for Chapter 1, the staff stated that it would require that applicants for FDA/DC specify the specific ANSI standard covering this subject. In Revision 4, EPRI revised Section 5.3.5 to require that designers provide recommendations and limits, including references to specific ANSI standards, if any, for each product evaluated for use in the NSSSs and other applications in ALWRs. This revision addresses the staff's concern identified in the DSER and is acceptable.

### 5.3.6 Electrical Materials

Section 5.3.6 of Chapter 1 requires that materials used in the plant's electrical systems, particularly those used in safety-related applications, be reviewed 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 subsequent consequences should a fire occur.

In the DSER for Chapter 1, the staff concluded that this requirement was acceptable. However, it was not clear if it was sufficient to address such issues as aging of cable insulations and other electrical materials over the design life and full range of environmental conditions. In the DSER, the staff stated that it would evaluate this issue during its review of an individual application for FDA/DC. In Revision 4, EPRI revised Section 5.3.6 to require that materials used in the plant's electrical systems, particularly those used in safety-related applications, be reviewed for functional reliability over the design life and operating and accident conditions, including effects of aging. The staff concludes that it is acceptable. However, additional staff review of aging of cable insulation is necessary. Therefore, the staff will ensure that vendors have adequately addressed the issue of aging of cable insulations and other electrical materials 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 (AWS) specification as appropriate for code or non-code construction.

These requirements are in conformance with the requirements of ASME Code, Section III. Therefore, the staff considers them acceptable.

5.4 Process Controls

5.4.1 Surface Condition

Section 5.4.1 of Chapter 1 of the passive Requirements Document requires that the effect of fabrication and installation processes that adversely affect the surface condition or microstructure (e.g., forming, bending, welding, heat treating, 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 considers this requirement acceptable because installation processes will be controlled through all stages, which will ensure 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. At a minimum, this control will be applied to the fabrication steps listed below for the materials and processes involved.

Ferritic Steels

Preheat and post-weld heat treatment of ferritic steels will be controlled in accordance with RG 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel." Applications in areas of limited accessibility will be qualified in accordance with RG 1.71, "Welder Qualifications for Areas of Limited Accessibility." Heat inputs in cladding will be in accordance with RG 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 RG 1.44, Control of Use of Sensitized Stainless Steel." Interpass temperatures will be controlled to improve resistance to IGSCC. Stainless steels will be welded in

accordance with RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Material," and applications in areas of limited accessibility will be qualified in accordance with the guidance of RG 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 considers these requirements acceptable because they are in accordance with the recommended practices in RGs 1.31, 1.50, and 1.71.

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. At 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.

- Where inservice UT inspection of a component or a weld will be required, the requirements on the design, materials, and fabrication inspections for that component or weld will ensure that adequate UT inspection in the field can be performed.

The staff considers these requirements acceptable because they are in accordance with the requirements of ASME Code, Section III, and RGs 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.

The staff considers this requirement acceptable because it complies with the guidance given in SRP Section 5.2.3, "Reactor Coolant Pressure Boundary Materials."

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.

The staff considers this requirement acceptable because it is in accordance with accepted industry practice.

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 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 considers this requirement acceptable because is in accordance with accepted industry practice.

5.5 Environmental Conditions

5.5.1 Range of Environmental Conditions

Section 5.5.1 of Chapter 1 of the passive 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 the selection of ALWR materials. These environmental conditions will be consistent with the specific requirements in other chapters of the passive Requirements Document.

The staff considers these requirements 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 1.5-3 in this chapter of structural materials resistant to IASCC and IGSCC are not available for RPV boundary and non-replaceable interval components, or in combination with resistant materials and residual stress limitations (in the case of IASCC).

The specific HWC control values in EPRI's guidelines relative to recirculating piping (e.g., 230 milliavolts) 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 RCS 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 the result of HWC. The evaluation will include erosion/corrosion, radiation buildup, and pitting at shutdown. The staff will evaluate HWC use during its review of an individual application for FDA/DC to determine what advances have been made in this area.

The staff considers the requirements for BWR water chemistry acceptable. Their implementation will ensure that the BWR water chemistry is compatible with the selected materials.

Table 1.3 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($\mu$S/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:

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

Reactor water cleanup system 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.

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 1.4 in this chapter.

Table 1.4 PWR Water Chemistry Guidelines

| Water Quality Parameter | MWST | MWSG |
|---|---|---|
| pH (minimum) | - | 7.0 |
| (maximum) | - | 7.5 |
| Conductivity at 25 °C ($\mu$S/cm) | 0.2 | 0.1 |
| Sodium (ppb) | - | <3.0 |
| Silica (ppb) | - | 10 |
| Oxygen (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.

Note:

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

The staff considers these requirements acceptable. Their implementation will ensure that the PWR water chemistry is compatible with the selected materials. However, in the DSER for Chapter 1, the staff stated that Section 5.5.2.4 of Chapter 1 of the passive Requirements Document should reference EPRI NP-7077, Revision 2, instead of EPRI NP-5960, Revision 1. This was identified as an open issue. In Revision 4, EPRI revised Section 5.5.2.4 of Chapter 1 to specify the use of EPRI NP-7077, Revision 2. Therefore, this open issue is closed.

## 5.6 Conclusion

The staff has reviewed the general materials, materials fabrication, and water chemistry requirements in Section 5 of Chapter 1 of the passive Requirements Document. These requirements supplement the regulatory guidance in the applicable SRP sections and, therefore, are acceptable. Their implementation will ensure that the affected nuclear power plant components will perform in service as designed.

# 6 RELIABILITY AND AVAILABILITY

Section 6 of Chapter 1 of the passive Requirements Document gives reliability and availability requirements for a design reliability assurance program (D-RAP) and provides the supporting rationale for each requirement. The EPRI D-RAP encompasses SSCs whose reliability has a significant effect on core damage frequency or whose reliability has a significant effect on challenges to emergency core cooling systems or plant availability for power production. In Section 6.1, EPRI addresses D-RAP goals and objectives. Section 6.2 includes the basic program elements of the D-RAP. Section 6.3 addresses the D-RAP performance standards. Section 6.4 discusses the aspects of systems design. The staff has reviewed Section 6 of Chapter 1 of the passive Requirements Document through Revision 6.

## Review Criteria

The need for a safety-oriented reliability effort for the nuclear industry was identified by the NRC in the Three Mile Island Action Plan (NUREG-0660) Item II.C.4, "Reliability Engineering." Subsequently, 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 action, 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 NRC concluded from this research 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.

The staff's position on RAP for ALWRs was developed as described in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," (Annex A to Appendix B of Chapter 1 of this report). The staff's final position on RAP was presented in SECY-94-084, "Policy and Technical Issues Associated With The Regulatory Treatment of Non-Safety Systems in Passive Plant Designs" (Annex B to Appendix B of Chapter 1 of this report).

SECY-94-084 contained the following applicable regulation for D-RAP:

An application for design certification or for a combined license must contain:

(a) the description of the reliability assurance program used during the design that includes scope, purpose, and objectives;

(b) the process used to evaluate and prioritize the structures, systems, and components in the design, based on their degree of risk-significance;

(c) a list of structures, systems, and components designated as risk-significant; and

(d) for those structures, systems, and components designated as risk-significant:

(i) A process to determine dominant failure modes that considered industry experience, analytical models, and applicable requirements; and

(ii) Key assumptions and risk insights from probabilistic, deterministic and other methods that considered operation, maintenance and monitoring activities.

Each COL applicant that references a passive ALWR design must implement the D-RAP approved by the NRC.

The Staff Requirements Memorandum (SRM) dated June 30, 1994, regarding SECY-94-084, Section E, Reliability Assurance Program (RAP) approved an applicable regulation for D-RAP. The SRM also stated that the staff should ensure that the objectives of the operational reliability assurance program (O-RAP) are incorporated into existing programs for maintenance or quality assurance.

This safety evaluation reflects the position described in SECY-94-084, as modified by the June 30, 1994, SRM. The staff limited its review of Section 6 of the passive Requirements Document to the development and implementation of the D-RAP during design activities conducted by FDA/DC or COL applicants.

Scope of Evaluation

The passive Requirements Document primarily addresses D-RAP requirements; only a limited number of requirements in the passive Requirements Document apply to plant operations. These requirements include O-RAP objectives which should be incorporated into existing programs for maintenance and quality assurance by an individual applicant for a COL.

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 and outage duration goals that are specified in the passive 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 a COL applicant will be required to provide additional information or guidance. For requirements of the second type, the staff has provided clarification regarding what additional information a plant designer or a COL applicant should provide when making a submittal that references the passive Requirements Document.

In its DSER for Volume III of the passive Requirements Document, the staff identified five open issues associated with (1) definition of the basic framework (scope, purpose, and objective) and the program elements of a RAP, and description of how the elements would be applied to the plant structures, systems, and equipment; (2) description of overall reliability and availability design goals based on safety requirements that have core-damage

frequencies or probabilities associated with them; (3) description of reliability engineering techniques used during the design phase to ensure that overall design reliability and availability goals are met; (4) specification of the connection of the system requirements to overall safety, reliability, and availability goals; and (5) specification of the rationale for the designer to develop a reliability program for safety-related systems and to implement a reliability program for non-safety-related systems.

In responding to the DSER open issues, EPRI made significant changes to Section 6 of Chapter 1. The staff has reviewed EPRI's revisions and concludes that they do not conflict with the NRC position on RAP as described in SECY-94-084 and the June 30, 1994, SRM. These changes resolve all of the five DSER open issues listed above. The details of the staff's evaluation are presented in Sections 6.1 through 6.4, below.

## 6.1 D-RAP Goals and Objectives

Section 6.1 of Chapter 1 of the passive Requirements Document contains the D-RAP goals and objectives.

A portion of the open issue concerning the definition of the basic framework (scope, purpose, and objective) of a RAP, was closed by the D-RAP goals and objectives provided by EPRI in revisions to Section 6.1. The remainder of the open issue, definition of the program elements and a description of how the elements would be applied to the plant SSCs, was resolved by EPRI's revisions to Section 6.2, as described below.

The staff concludes that the EPRI-proposed requirements in Section 6.1 do not conflict with the NRC position on RAP as described in SECY-94-084 and the June 30, 1994, SRM and are, therefore, acceptable.

## 6.2 Basic Program Elements of the D-RAP

Section 6.2 of Chapter 1 of the passive Requirements Document describes the basic program elements that EPRI requires to be included in the D-RAP.

In the DSER, the staff concluded that EPRI should specify the rationale for the designer to develop a reliability program for safety-related systems and to implement a reliability program for non-safety systems. In Revision 3, EPRI revised Section 6 to satisfactorily resolve this open issue by removing the differentiation between safety-related and non-safety-related systems. The staff concluded that these changes are acceptable, therefore, this open issue is closed.

In the DSER, the staff identified an open issue concerning the definition of the basic framework (scope, purpose, and objective) of a RAP and program elements and how the elements would be applied to the plant systems, structures, and components (SSCs). In Revision 3, EPRI revised Section 6.1 by adding requirements for goals and objectives that define the basic framework of a RAP and program elements. EPRI also revised Section 6.2, by describing how the program elements would be applied to the plant SSCs. The staff concludes that these changes are acceptable, therefore, this open issue is closed.

In the DSER, the staff also identified an open issue concerning the overall quantitative reliability and availability design goals based on safety requirements that have core damage frequencies or probabilities associated with them. The staff concludes that the analysis requirements, reliability activity task descriptions, and control of PRA design assumptions provided in Section 6.2 are acceptable. Therefore, this open issue is closed.

In the DSER, the staff concluded that a RAP should contain the reliability engineering techniques used during the design phase to ensure that overall design reliability and availability goals are met. The basic program elements including analysis requirements, reliability activity task descriptions, and control of PRA design assumptions provided in Section 6.2 satisfactorily resolved a portion of the open issue concerning reliability engineering techniques. The remainder of this open issue was satisfactorily resolved by the requirements of Sections 6.3 and 6.4. Therefore, this open issue is closed.

The staff concludes that the EPRI-proposed requirements in Section 6.2 do not conflict with the NRC position on RAP as described in SECY-94-084 and the June 30, 1994, SRM and are, therefore, acceptable.

The staff's detailed evaluation of Section 6.2 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 consistent with the SSC reliability and availability assumptions of 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 passive 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.

Sections 6.2.1.1 through 6.2.1.8 contain the following analysis requirements: Quantitative Requirements, Development of System Level Models, Modeling Techniques, LWR Experience Reviews, Unavailability Allocation, Content of Analysis, Documentation of Analysis, and the Design Organization Interface. The staff concludes that the analysis requirements do not conflict with existing NRC requirements or with the NRC position on RAP as described in SECY-94-084 and the June 30, 1994, SRM and are, therefore, acceptable.

The staff also concludes that, in addition to PRA methodology, deterministic or other methods may be used for making decisions about the adequacy of plant system designs. Furthermore, the staff encourages a vendor or COL 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 passive Requirements Document describes analytical methods for the man-machine interface systems (M-MIS) design. The staff concludes that these or similar methods should be used for all reliability and maintainability analyses performed by the plant designer or COL applicant. The staff will evaluate this issue during its review of an individual application for FDA/DC or a COL.

Design Organization Interface

Section 6.2.1.8 of Chapter 1 requires that plant designers identify the design organization responsible for coordinating D-RAP activities with the plant owner's organization and the formal communication channels for resolving D-RAP issues. The staff concludes that, in addition to the EPRI requirements in this section, plant designers and COL applicants 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 passive Requirements Document specifies requirements that should ensure that the M-MIS design is coordinated and implemented properly. An application for FDA/DC and COL should contain an organizational description and accountability for the RAP in similar depth and detail as that contained in Section 3.1.2 of Chapter 10 of the passive Requirements Document. The staff will evaluate this description during its review of an individual application for FDA/DC and a COL.

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 the NRC position on RAP as described in SECY-94-084 and the June 30, 1994, SRM and are, therefore, acceptable.

Plant Reliability Data System Engineering Database

Section 6.2.3 of Chapter 1 requires the plant designer to supply the initial input information necessary to establish an engineering database for all SSCs considered to be candidates for the D-RAP. The engineering database will provide sufficient engineering data for D-RAP implementation. The engineering data will be provided in a format that allows direct input into the database without editing, translating, or other manual processing of the data. The staff concludes that the EPRI-proposed requirements do not conflict with the NRC position on RAP as described in SECY-94-084 and the June 30, 1994, SRM and are, therefore, acceptable.

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 do not conflict with the NRC position on the RAP as described in SECY-94-084 and the June 30, 1994, SRM and are, therefore, acceptable.

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 passive 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 the NRC position on RAP as described in SECY-94-084 and the June 30, 1994, SRM and are, therefore, acceptable.

## 6.3 D-RAP Performance Standards

Section 6.3 of Chapter 1 of the passive Requirements Document contains the quantitative reliability and availability performance standards for use in the design phase.

In the DSER, the staff concluded that a RAP should contain the reliability engineering techniques performed during the design phase to ensure that overall design reliability and availability goals are met. Because EPRI had not described how these techniques will relate to the overall reliability program and how they will be integrated and considered in the RAP, this was identified as an open issue. In Revision 3, EPRI revised Section 6.3 to provide acceptable quantitative reliability and availability performance techniques for use in the design phase and described how these techniques are to be integrated and considered in the RAP. The remainder of this open issue was satisfactorily resolved by revisions to the requirements of Section 6.2 and 6.4. Therefore, this open issue is closed.

In the DSER, the staff identified an open issue regarding the relationship of the system requirements to overall plant safety, reliability, and availability goals. In Revision 5, EPRI revised Section 6.3 to clarify that the design features in that section will be consistent with ALWR core damage frequency and consequences goals. Therefore, this open issue is closed.

The staff has evaluated the sub-sections of Section 6.3, addressing plant availability and outage durations as they affect economic considerations, only with regard to the general relationship between safety and economic requirements.

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 seismic and 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, the staff will not use EPRI's public safety 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, an applicant's submittal for FDA/DC should include apportionment of these contributions to the core damage frequency. However, this apportionment is an aspect of the industry's design process only and, as stated in the June 30, 1994, SRM, will not be effectively translated into new regulatory requirements which result in core damage frequency and containment conditional failure probability values that are lower than the subsidiary objectives which the Commission has approved for use.

Inadvertent Depressurization

Section 6.3.2 of Chapter 1 states that non-safety-related active reactor coolant system 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 concludes that these requirements do not conflict with existing NRC requirements or with the NRC position on RAP as described in SECY-94-084 and the June 30, 1994, SRM and are, therefore, acceptable.

Station Blackout

Section 6.3.3 of Chapter 1 requires plant designers to provide design features to ensure that the frequency of station blackouts exceeding 2 hours duration can be demonstrated by PRA methods to have an expected frequency of less than 10E-3 event per reactor-year. The staff concludes the requirements of this section do not conflict with existing regulatory requirements or with the NRC position on RAP as described in SECY-94-084 and the June 30, 1994, SRM and are, therefore, acceptable.

Trip Frequency

Section 6.3.4 of Chapter 1 states the plant designer will perform an evaluation of potential system failures that will result in reactor trips that show the expected trip frequency to be less than one per year. Because human errors cause a large fraction of reactor trips, emphasis will be placed on the human factors aspects of operation and surveillance testing. The staff concludes the requirements of this section do not conflict with existing regulatory requirements or with the NRC position on RAP as described in SECY-94-084 and that June 30, 1994, SRM and are, therefore, acceptable.

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 compromise plant safety. The staff concludes that this requirement does not conflict with existing regulatory requirements or with the NRC position on RAP as described in SECY-94-084 and the June 30, 1994, SRM and is, therefore, acceptable.

Forced Outages

Section 6.3.6 of Chapter 1 states the plant will be designed so that the frequency and duration for forced outages will not exceed 5 days per year. The staff's review of this requirement was limited to areas where economic considerations could compromise plant safety. The staff concludes that this requirement does not conflict with existing regulatory requirements or with the NRC position on RAP as described in SECY-94-084 and the June 30, 1994, SRM and is, therefore, acceptable.

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's review of this requirement was limited to areas where economic considerations could compromise plant safety. The staff concludes that this requirement does not conflict with the NRC position on RAP as described in SECY-94-084 and the June 30, 1994, SRM or with existing NRC requirements, and is, therefore, acceptable.

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's review of this requirement was limited to areas where economic considerations could compromise plant safety. The staff concludes that this requirement does not conflict with the NRC position on RAP as described in SECY-94-084 and the June 30, 1994, SRM or with existing NRC requirements, and is, therefore, acceptable.

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's review of this requirement was limited to areas where economic considerations could compromise plant safety. The staff concludes that this requirement does not conflict with the NRC position on RAP as described in SECY-94-084 and the June 30, 1994, SRM or with existing NRC requirements, and is, therefore, acceptable.

## 6.4 System Design

Section 6.4 of Chapter 1 of the passive Requirements Document describes the qualitative processes to be used in system design.

In the DSER, the staff concluded that EPRI should provide the reliability engineering techniques used during the design phase to ensure that overall design reliability and availability goals are met. This was identified as an open issue. The staff concludes that EPRI's revised requirements for the qualitative processes to be used in system design provided in Section 6.4 are acceptable. The remainder of this open issue was satisfactorily resolved by the requirements of Section 6.2 and 6.3. Therefore, Open Issue P.1.0-14 is closed.

The staff concludes that the EPRI-proposed requirements in Section 6.4 do not conflict with any NRC requirements or with NRC position on RAP as described in SECY-94-084 and the June 30, 1994, SRM and are, therefore, acceptable. The staff's detailed evaluation is given below.

Shared Systems

Section 6.4.1 of Chapter 1 requires that, for multiple-unit plants on a single site, the number of shared systems be limited to auxiliary support systems such as sewer, auxiliary steam, or site security. For any systems that are shared, the plant designer or COL 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 the NRC position on RAP as described in SECY-94-084 and the June 30, 1994, SRM or with existing NRC requirements, and are, therefore, acceptable.

Startup and Testing

Section 6.4.2 of Chapter 1 requires that the plant designer or COL applicant review and optimize the startup testing program for initial startup and for startup following refueling and 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 the NRC position on RAP as described in SECY-94-084 and the June 30, 1994, SRM or 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 described in Section 6.3 of Chapter 1. The staff concludes that these requirements do not conflict with the NRC position on RAP as described in SECY-94-084 and the June 30, 1994, SRM or 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 the NRC position on RAP as described in SECY-94-084 and the June 30, 1994, SRM or 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 the NRC position on RAP as described in SECY-94-084 and the June 30, 1994, SRM or 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 passive 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 COL applicant. Additional guidance and requirements will be necessary to support the completion of a COL applicant's D-RAP (site-specific design information) the incorporation of O-RAP objectives into existing programs for maintenance and quality assurance.

The staff concludes that the reliability and availability requirements in Section 6 of Chapter 1 of the passive Requirements Document are consistent with accepted industry practices and principles, are responsive to the staff's DSER open issues, does not conflict with the NRC position on the RAP as described in SECY-94-084 and the Commission's June 30, 1994, SRM, or with existing regulatory requirements and guidelines. Therefore, they are acceptable.

However, by themselves, the reliability and availability requirements do not provide sufficient information for the staff to determine if the ALWR design referencing the EPRI requirements will adequately incorporate RAP considerations in a manner that will ensure plant safety and reliability. Therefore, an applicant referencing the passive Requirements Document will be required to provide sufficient information to demonstrate that its RAP will result in a plant that is designed and will perform in a manner that will ensure plant safety and reliability.

# 7 CONSTRUCTION AND CONSTRUCTIBILITY

Section 7 of Chapter 1 of the passive Requirements Document specifies minimum requirements and goals proposed by EPRI with regard to the construction of an ALWR passive plant. EPRI states that these requirements support the overall objectives described in Volume I of the EPRI 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, the staff did identify items that are part of its regulatory responsibility. Its evaluation of these follows.

## 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 owners/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 a specific application for a COL.

## 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 dated September 1987 for Section 7 of Chapter 1 of the Requirements Document for evolutionary plants, the staff identified an open issue that construction activities for which the owner/builder must provide a site organization should include quality assurance/quality control. This issue resulted from the staff's review of Revision 0 of the Requirements Document.

In response to the staff's concern, EPRI revised Section 7.2.8.2 of Chapter 1 of the Requirements Document for both the evolutionary and passive plants to make it clear that the site organization plan must include provisions for quality control and quality assurance. In its DSER for Chapter 1, the staff concluded that this revision resolved its original concern. However, EPRI had added additional information that was not clear with respect to EPRI's intent

regarding quality assurance. For example, Section 7.2.8.2 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, refer to the new maintenance rule, 10 CFR 50.65(b)(2)). The staff concluded that pertinent quality assurance provisions should be applied to these activities and items and stated that this issue was part of the overall issue regarding the regulatory treatment of non-safety systems (RTNSS) for passive plants, discussed in the regulatory departure analysis of SECY-94-084, which is provided in Annex B of Appendix B to this chapter. This was identified as an open issue. As described in SECY-93-087, the staff will review individual applications for FDA/DC to ensure that quality assurance provisions are applied commensurate with an item's or activity's importance to safety. Therefore, this open issue is closed.

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 of the passive Requirements Document, 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 first fuel loading, but that this milestone was beyond the scope of the ALWR passive Requirements Document. This will be addressed by the staff during its review of a specific application for a COL.

The staff expects that at least 60 days before fuel is loaded, a licensee for a COL 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 entirety as they will when the reactor is operating and will remain so. The COL determination that operational status has been achieved must be based on tests conducted under realistic operating conditions of sufficient duration that demonstrated 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 the 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 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. This will be addressed by the staff during its review of a specific application for a COL, should the applicant propose use of these techniques.

# 8 OPERABILITY AND MAINTAINABILITY

## 8.1 Introduction

In Section 8 of Chapter 1 of the passive 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 passive 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.

### Radiation Exposure

To reduce worker radiation exposure, EPRI states that temperature and humidity in radiation areas where work is to be performed will be maintained at a level required for worker efficiency. 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 reasonably achievable (ALARA) and, therefore, are acceptable.

### Personnel Qualification

In Section 8 of the DSER dated September 1987 for Chapter 1 of the Requirements Document for evolutionary plants, the staff stated its concern that EPRI had indicated that it would add a statement to the Requirements Document regarding the qualification of plant operating personnel. Section 8.2.6.5 of Chapter 1 of the passive Requirements Document indicates that these personnel will meet the requirements of ANSI/ANS 3.1, "Selection, Qualification and Training for Nuclear Power Plants." The staff concludes that this is consistent with the regulatory guidance in SRP Sections 13.2.1, "Reactor Operator Training," and 13.2.2, "Training for Non-Licensed Plant Staff," and is, therefore, acceptable.

Personnel and Staffing

In its letter dated May 5, 1991, EPRI stated that qualification and training requirements of the plant operating personnel are the responsibility of the plant owner and are outside the scope of the passive Requirements Document. Section 8.2.6.5 of Chapter 1 includes a requirement that the plant operating organization be staffed by personnel who meet the requirements of ANSI/ANS 3.1. Since 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).

Human Factors Considerations

In Section 8.2.B.4 of the DSER for Chapter 1 of the Requirements Document for evolutionary plants, the staff recommended that IEEE P1023/D5, "Guide for the Application of Human Factors Engineering to Systems, Equipment, and Facilities of Nuclear Power Generating Stations," and EPRI-NP-2360, "Human Factors Methods for Assessing and Enhancing Power Plant Maintainability," be referenced in this section. In the DSER for Chapter 1 of the passive Requirements Document, the staff concluded that this recommendation was applicable to passive plant designs. However, these documents were not referenced in the passive Requirements Document. Therefore, this was identified as an open issue. In Revision 4, EPRI changed Section 8.2.1.2 of Chapter 1, to reference IEEE 1023-1988, "Guide for the Application of Human Factors Engineering to Systems, Equipment, and Facilities of Nuclear Power Generating Stations." Also, in its letter dated September 29, 1992, EPRI stated that Section 3.7.7 of Chapter 10 contains a reference to EPRI NP-4350, "Human Engineering Design Guidelines for Maintainability," which in turn references the older report EPRI NP-2360, "Human Factors Methods for Assessing and Enhancing Power Plant Maintainability." This revision and clarification are acceptable; therefore, this open issue is closed.

60-Year Life

Section 8.2.3.3 of Chapter 1 of the passive Requirements Document 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 passive 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 limitation is specified in the Atomic Energy Act and NRC regulations.

## 8.3 Minimizing Dose Levels to Personnel (ALARA)

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

To achieve its 100 person-rem/year dose goal 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 to facilitate rapid disassembly/assembly for inspection and maintenance; and the design of heating and ventilation systems to control temperature and humidity in radiation work areas.

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 and steam generator channel heads) where this treatment will significantly reduce the dose to personnel during maintenance.

The plant designer will perform an analysis to determine the effectiveness of using robotic applications in the ALWR. Candidate inspection and surveillance functions for robotic application include reading of instruments and gauges, performing radiation surveys and measuring radiation levels, and taking smear surveys. Candidate maintenance functions 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 passive 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 consideration of replacing a security officer are outside the scope of the passive Requirements Document. This will be addressed by the staff during its review of a specific application for a COL.

These design features are intended to minimize dose levels to plant personnel, are in compliance with RG 8.8, and are, therefore, acceptable.

## 8.4 Facility Requirements

Section 8.4 of Chapter 1 of the passive 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.

This section includes requirements for the design of the plant's personnel access portal (PAP). The security requirements in Section 8.4 are supplemented by those in Section 5 of Chapter 9 of the passive Requirements Document.

EPRI states that the ALWR will be designed to allow controlled personnel access to the plant for the necessary 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 anti-contamination 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 are not in 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 passive Requirements Document specifies requirements to facilitate the replacement of all major components other than the reactor vessel and basic plant structures. The plant designer will also include plans for 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 are not in conflict with the Commission's regulations and guidance and are, therefore, acceptable.

8.6 Inspection and Testing

Section 8.6 of Chapter 1 of the passive Requirements Document specifies general requirements for inservice inspection features and techniques to enable the performance of such testing in accordance with Section XI of the ASME Code. The staff concludes that these general requirements are not in conflict with the Commission's regulations and guidance. The staff's detailed review of inspection and testing is presented in other chapters throughout this report.

8.7 Hazardous and Toxic Chemicals

Section 8.7 of Chapter 1 of the passive Requirements Document specifies that the use of hazardous and toxic chemicals within 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 are not in conflict with the Commission's regulations and guidance and are, therefore, acceptable.

8.8 Conclusions

The staff concludes that the requirements of Section 8 of Chapter 1 of the passive Requirements Document are not in conflict with the Commission's regulations and are, therefore, acceptable.

# 9 QUALITY ASSURANCE

Section 9 of Chapter 1 of the passive Requirements Document identifies the major elements of an overall quality assurance (QA) program for a passive 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 a passive 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 Section 9 of the DSER for Chapter 1 of the Requirements Document for evolutionary plants, the staff expressed its concern that EPRI's responses to earlier staff comments had not been incorporated into the Requirements Document. Sections 9.2.3 and 9.2.4 of Chapter 1 of the passive Requirements Document incorporate the requested changes. The staff concludes that this section is in accordance with the regulatory guidance in the SRP and RGs as described below and is, therefore, acceptable.

In Section 9 of the DSER for Chapter 1 of the Requirements Document for evolutionary plants, the staff stated that two line items should be added to Table 9-1 of the Requirements Document. Table 9-1 of Chapter 1 of the passive Requirements Document incorporates the requested changes. The staff concludes that this is, therefore, acceptable.

In Section 10 of the DSER for Chapter 1 of the Requirements Document for evolutionary plants, the staff stated that the cross-references in Table B.1-1 were incorrect. As discussed 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; 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 passive Requirements Document. Table B.1-2 also lists RGs 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). Except as noted below, 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 passive Requirements Document. In the DSER for Chapter 1, the staff concluded that these references were acceptable. However, the commitment to RGs 1.26 and 1.29 was shown in Table B.1-2 as an optimization subject regarding the BWR main steam isolation valve (MSIV) leakage control system, with Chapter 3 of the passive Requirements Document being listed as the lead chapter. The staff concluded that Table B.1-2 should clearly state that both of these RGs will be complied with, with the exception of the MSIV leakage control system. The criteria to be followed for this system are discussed by the staff in the regulatory departure analysis of SECY-93-087 contained in Annex A of Appendix B to this chapter. The staff stated that EPRI had committed to clarify the opening section of Appendix B to Chapter 1 of the passive Requirements Document so that the issue could be closed and that it would confirm the resolution of this issue when it received the applicable revision to the passive Requirements Document. Instead, in its letter dated September 29, 1992, EPRI provided clarification that both of these RGs concern an optimization issue associated with BWR MSIVs and leakage

control, as described in Section 2.3.1 of Appendix B to this chapter and noted in Table B.1-2. The staff concludes that this response is acceptable; therefore, this confirmatory issue is closed.

In Section 6.1.2 of the DSER for Chapter 10 of the Requirements Document for evolutionary plants, 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 Requirements Document for evolutionary plants, EPRI appears to limit its QA commitment to safety-related items. In the DSER for Chapter 1 of the passive Requirements Document, the staff stated that it was concerned that the document could be interpreted to not require a commitment to apply pertinent QA program requirements to software, facilities, structures, systems, and components that have importance to safety even though they are not safety related. The staff concluded that pertinent QA provisions should be applied to these activities and items commensurate with their importance to safety.

The staff stated that this issue was part of the overall issue regarding the regulatory treatment of non-safety systems for passive plants that was discussed in the regulatory departure analysis of the DSER for Appendix B to this chapter. This was identified as an open issue. As described in SECY-93-087, which is provided in Annex A of Appendix B to this chapter, the staff will review individual applications for FDA/DC to ensure that QA provisions are applied commensurate with an item's or activity's importance to safety. Therefore, this open issue is closed.

Conclusion

The staff concludes that the requirements of Section 9 of Chapter 1 of the passive Requirements Document are not in conflict with the Commission's regulations and are, therefore, acceptable.

# 10 LICENSING

Section 10.1 of Chapter 1 of the passive Requirements Document specifies EPRI-proposed licensing requirements for future ALWRs. EPRI stated that Section 1 of Appendix B to Chapter 1 contains a list of NRC regulations and guidelines currently applicable to LWR design and identifies EPRI's 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 Requirements Document when that document is completed.

Section 10.2 of Chapter 1 originally stated 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 passive Requirements Document. EPRI stated that these requirements and guidance included applicable Commission regulations specified in Title 10 of the Code of Federal Regulations, general design criteria, NRC policy statements, RGs, the SRP, and other documentation that resolves unresolved and generic safety issues. In the DSER for Chapter 1, the staff stated that although it understood EPRI's need to "freeze" the requirements it addresses to those in effect on January 1, 1990, the staff expected that the FDA/DC applications would be in compliance with the Commission's regulations and guidance that were applicable and in effect at the time FDA/DC 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 passive Requirements Document may be developed as the staff completes its reviews of the detailed design information provided in the FDA/DC applications, and as these designs are litigated in the FDA/DC hearings. Therefore, the staff expects that the ALWR plant designers will comply with issue resolutions adopted by the NRC staff during its reviews of the FDA/DC applications 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 Section 10 of the DSER for Chapter 1 of the Requirements Document for evolutionary plants, the staff stated that the cross-references in Table B.1-1 were incorrect. EPRI has corrected this table and has committed to finalize it when it completes the Requirements Document.

Section 10.2.4 of Chapter 1 of the passive Requirements Document 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 compliance 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.

- Comply With Intent

  The "comply with intent" designation indicates that the ALWR design will comply with the intent of the regulatory items where a fundamentally different design approach has been taken to satisfy the objective of regulatory requirements and guidance items. Each designated item is discussed in Section 4 of Appendix B of Chapter 1 of the passive Requirements Document.

- 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 passive Requirements Document.

Section 10.2 of Chapter 1 states that the plant designer will provide a design that is consistent with Sections 2, 3, and 4 of Appendix B to Chapter 1 of the passive Requirements Document. These include EPRI-proposed resolutions for optimization subjects, generic safety issues, and regulatory items designated as "comply with intent," respectively.

In Section 10 of the DSER for Chapter 1 of the Requirements Document for evolutionary plants, 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. 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 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 as a substitute for any portion of the staff's review of an individual application for FDA/DC.

Inspections, Tests, Analyses, and Acceptance Criteria

Section 52.47(a)(1)(vi) of 10 CFR requires the design certification applications to include proposed tests, inspections, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the tests, inspections, 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 COL, 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 on ITAAC scope and content. In its letter dated May 13, 1991, EPRI indicated that it did not plan to provide ITAAC guidance, but was supporting NUMARC's development of industry guidance on ITAAC for design certification. Each plant designer will submit ITAAC for its specific design for NRC review using the industry guidance.

In the design certification process, the design detail to be developed by an applicant for FDA/DC can be embodied in three parts: Tier 1 information that will be submitted in the application and certified by rulemaking, Tier 2 information that will be submitted in an application but not certified, and the information not submitted but available for NRC audit. This two-tiered approach is to standardize design details to the maximum extent practical, 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 detail design phases, such as system and key component descriptions, 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 on the scope and content of ITAAC and is evaluating pilot ITAAC submittals based on the General Electric Company Advanced BWR 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. While including appropriate guidance on scope and content for ITAAC submittals in the passive Requirements Document would ensure that each design submittal provides a complete and adequate ITAAC package for staff review, ITAAC is clearly the responsibility of the plant designer during the design certification phase. The staff will evaluate the proposed ITAAC during its review of an individual application for FDA/DC and for a COL.

# 11 DESIGN PROCESS

## 11.1 Introduction

In Section 11 of Chapter 1 of the passive Requirements Document, EPRI proposes requirements for the design process of an ALWR plant. 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, 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 as 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 in similar operating environments and applications. Although the passive 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. The staff discusses the process that it will use for determining the need for a prototype or other demonstration facility for the ALWR in the regulatory departure analysis of SECY-93-087, contained in Appendix B to this chapter. In addition, in this analysis, the staff discusses the necessity for separate effects analysis and scaled integral testing for the passive plant designs.

## 11.3 Design Life

Section 11.3 of Chapter 1 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 regulations.

## 11.4 Plant Simplification

Section 11.4 of Chapter 1 states that the ALWR design should 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 implementation of this objective is 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 states that the ALWR design will be developed as a standard plant design, including, at 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 issues of the Commission's standardization policy, and the staff's efforts in the review of the EPRI Requirements Document. The staff will ensure that implementation of this concept is in accordance with the Commission's regulations and guidance during its review of an individual application for FDA/DC.

11.6 Specific Requirements of the ALWR Design Process

Section 11.6 of Chapter 1 specifies that the plant designer will use design methods required by applicable standards, codes, and regulations, as well as 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 passive Requirements Document.

The staff discusses the applicable codes and standards to be used in the ALWR designs in the regulatory departure analysis of SECY-93-087, contained in Appendix B to this chapter.

11.7 Project Information Network

Section 11.7 of Chapter 1 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 database, systematic retrieval of information, standard identification of data, and provision of a single source of technical data. The staff evaluates the PIN and its relationship to the information management system that EPRI specifies in Section 11.12 of this chapter.

11.8 Design Development Plan

Section 11.8 of Chapter 1 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.

11.9 Configuration Management

Section 11.9 of Chapter 1 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.

The information addressed in 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 about protection of safeguards information and is, therefore, acceptable. These requirements do not conflict with the Commission's regulations and guidance.

11.10 Design Integration

Section 11.10 of Chapter 1 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. The staff concludes that these requirements do not conflict with the Commission's regulations and guidance.

11.11 Interdisciplinary Design Reviews

Section 11.11 of Chapter 1 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 of a wide range of experts will help to uncover potential problems during the design stage.

Many disciplines are specifically mentioned as requiring membership in the review group, but experts in physical security are not. 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 is impractical to list all possible specialties that will be required for the review. Nevertheless, in Revision 2, EPRI modified Section 11.11.2 to specify that additional special disciplines will be included as appropriate to the subject under review. The staff concludes that this change resolves its concern and is, therefore, acceptable. These requirements do not conflict with the Commission's regulations and guidance.

11.12 Information Management System

Section 11.12 of Chapter 1 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 designer 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 requirements for the IMS as well as the operational requirements.

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 2.

Section 5.9 of Attachment 1 to Section 11 of Chapter 1 originally stated: "The plant designer shall certify the IMS security system in accordance with Guidelines for Computer Security and Accreditation (GUIDE 83)." In the DSER for Chapter 1, the staff stated that GUIDE 83 did not appear to be a correct citation and identified this as an open issue. In Revision 4, EPRI revised Section 5.9 to require that the IMS security system be certified by the plant designer in accordance with National Bureau of Standards FIPS PUB 102, "Guidelines for Computer Security and Accreditation (GUIDE 83)." The staff concludes that this is acceptable. Therefore, this open issue is closed.

The staff discusses 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 staff concludes that these requirements do not conflict with the Commission's regulations and guidance.

11.13 Design/Construction Integration

Section 11.13 of Chapter 1 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 COL issuance were beyond the scope of the passive Requirements Document, it would delete the reference to rulemaking. The staff has verified that this change was made in Revision 2, and is acceptable. Item 2 in Attachment 2 summarizes in detail the design engineering required to support the construction schedule. Some of the detail design items listed in Item 2 have asterisks to indicate that parts of those items will be completed within the standard plant design, rather than depending on the plant owner to complete them. 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, in Revision 3, EPRI later elected to delete the security plan (layout and detail drawings) from Attachment 2 in response to 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.

The staff concludes that 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. This will be addressed by the staff during its review of a specific application for a COL.

## 11.14 Engineering Field Verification of As-Built Conditions

Section 11.14 of Chapter 1 states that the plant designer should identify and perform necessary engineering field verification activities to confirm the adequacy of the installation. These requirements do not conflict with the Commission's regulations and guidance.

## 11.15 Conclusions

The staff concludes that the requirements proposed by EPRI in Section 11 of Chapter 1 of the passive 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 passive Requirements Document specifies requirements for the mechanical equipment used in many systems of a nuclear plant.

## 12.2 Inservice Testing of Pumps and Valves

Sections 12.2.7.1 and 12.4.3.2 of Chapter 1 originally stated 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 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. The staff concluded that the ASME Code does not ensure the level of component operability that is desired for the evolutionary ALWR designs. In SECY-93-087, the staff concluded that this position is also applicable to the passive ALWR designs (see the regulatory departure analysis in Annex A of 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 the DSER for Chapter 1, the inservice testing of pumps and valves was identified as a policy issue for passive ALWRs for which the staff proposed a resolution similar to that for evolutionary plants. On April 2, 1993, the staff issued SECY-93-087, which contains staff recommendations relative to Policy Issue I.N. that are identical to those in SECY-90-016, which applies to evolutionary plant only. SECY-93-087 is provided as Annex A of Appendix B to Chapter 1 of this report. The staff's evaluation of EPRI's requirements in the passive Requirements Document relative to Policy Issue I.N. is discussed below. During its review of an individual application for FDA/DC, the staff will implement the positions identified in Sections 12.2.1 through 12.2.3 of Chapter 1 of this report. Therefore, this open issue is closed.

In its staff requirements memorandum on SECY-90-016, dated June 26, 1990, the Commission approved the staff's position to be used on evolutionary LWRs. In addition, the staff agreed with the recommendation of the ACRS that the guidelines of Generic Letter (GL) 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 Requirements Document for evolutionary plants. In these responses, EPRI committed to modify Section 12 of Chapter 1 of the Requirements Document to address the staff's concerns. However, in Section 3.2 of the DSER dated January 15, 1991, for Chapter 8 of the Requirements Document for evolutionary plants, 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 Requirements Document. In its letter dated May 17, 1991, the staff requested additional information on the passive Requirements Document. The staff noted in that letter that with certain exceptions, the request for additional information (RAI) is 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 2 to the passive Requirements Document and on EPRI's August 1, 1991, response.

12.2.1 Scope of the Inservice Testing (IST) Program

In response to the staff's request to clarify the term "essential" for the scope of the IST program for pumps and valves, EPRI stated that for its ALWR program, an "essential" component or system is viewed by the utilities as one that is important for safety, reliability, availability, and protecting their investment. EPRI further stated that inspection, maintenance, and inservice testing for essential equipment that is safety related is covered by the technical specifications. Inspection, maintenance, and testing for non-safety-related essential equipment will be based on a reliability program of predictive maintenance techniques designed to minimize challenges to passive safety-related systems. Predictive maintenance will involve appropriate inspections, testing, and preventive maintenance as determined by PRA insights. The requirements in Section 12 of Chapter 1 of the passive Requirements Document are for inspection and testing of non-safety-related (but essential) equipment. The passive Requirements Document now imposes the ASME Code testing requirements on all essential equipment. The staff concludes that this level of testing, at a minimum, is appropriate for the essential non-safety-related components. However, it concludes that there are uncertainties concerning the lack of a proven operational performance history for certain non-safety-related, but essential, pumps and valves. These uncertainties make the essential non-safety-related systems and components more important in providing the defense-in-depth to prevent and mitigate accidents and core damage. The staff will evaluate the plant designer's inservice testing requirements for the essential non-safety-related components in accordance with the RTNSS process as described in SECY-94-084. Therefore, the staff will evaluate this issue during its review of an individual application for FDA/DC.

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 passive 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.

In its letter dated May 17, 1991, the staff also requested that the ALWR reliability program be used to determine (1) the non-safety-related systems and components that will have to be relied on to meet the safety goals and (2) 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 are related to reliability and availability. In particular, Section 6.2.1 of Chapter 1 states that the plant designer will prepare an analysis that shows the adequacy of plant system's designs and supporting maintenance systems, recommended spare parts, surveillance tests, and test intervals needed to support SSC reliability and availability assumptions of the PRA. In the rationale portion of the same section, EPRI states 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 ensure 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 and important non-safety-related pumps and valves be provided early in the design phase. The staff discusses in detail the valve testability issue in Section 12.2.2 below. It discusses the overall guidelines for the reliability assurance program in Section 6 of this chapter.

12.2.2 Valves

In its letter dated May 17, 1991, the staff requested that the passive ALWR systems be designed to accommodate the applicable ASME Code, Section XI requirements for quarterly testing of pumps and valves, rather than allowing designs that only accommodate testing during cold shutdowns or refueling outages. In its letter dated August 1, 1991, EPRI responded to this request by stating that the frequency of testing should be determined by such factors as component design, application, PRA insights, and design alternatives. EPRI further stated that it did not propose to change the manner in which the code has been applied to recently licensed plants. For the reasons discussed in Section 12.2.1 above, the staff concludes that its position as stated in the May 17, 1991, letter will improve the component reliability for passive ALWRs. Revision 4 of the passive Requirements Document indicates that system design will include provisions to test essential pumps and valves in accordance with

the requirements of Section XI of the ASME Code. However, it does not specify that the systems will be designed to accommodate quarterly testing. ASME/ANSI OM-10, referenced in Section XI, ASME Code, 1989 Edition, provides for relaxation in the valve testing frequency from quarterly to cold shutdowns or refueling outages if testing during these plant conditions is not practical. However, these rules were written to address testing of valves in currently operating reactors where the detailed piping system designs were completed before promulgation of the inservice testing (IST) requirements and do not accommodate quarterly testing. The advanced passive reactors, for which the final design is not complete, have sufficient lead time available to include provisions in their piping system designs to allow testing at power. Quarterly testing is the base testing frequency in the code and the original intent of the code. Also, it may be necessary to test more frequently than during cold shutdowns or at every refueling outage to provide a component reliability commensurate with the importance of the safety functions to be performed and with system reliability goals. Therefore, to the extent practical, the passive ALWR piping systems should be designed to accommodate the applicable code requirements for quarterly testing of valves, rather than allowing designs that only accommodate testing during cold shutdowns or refueling outages. In its letter dated August 1, 1991, EPRI stated that the plants should be designed to meet code requirements and "[t]his includes the possibility to defer the testing in particular cases as permitted by the code." The staff position is that passive plants be designed to provide for quarterly valve testing. The provision in the code for deferring actual valve testing should only be used on a case-by-case basis when, for example, it is impractical to use the designed-in capability for quarterly testing. Because this staff position has not been clearly incorporated into the passive Requirements Document, the staff will evaluate this issue during its review of individual applications for FDA/DC.

In addition, the staff's position is that essential pumps should also be tested (in addition to just having the system designed to accommodate testing) in accordance with Section XI of the ASME Code, similar to that specified in Section 12.2.7.1 that is, "Essential valves shall be tested in accordance with applicable portions of Section XI of the ASME Code." For the above reasons, the staff concludes that EPRI's response is not acceptable. The staff will evaluate these positions during its review of individual applications for FDA/DC in accordance with the above position.

In its May 17, 1991, letter, the staff requested, at a minimum, a commitment to develop a program that will establish the frequency and the extent of disassembly and inspection of safety-related valves, including the basis for the frequency and extent of each disassembly. In Revision 4, EPRI provided requirements in Section 12.2.7.8 of Chapter 1 of the passive Requirements Document that are consistent with the staff request and are, therefore, acceptable.

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 passive Requirements Document were in compliance with the Commission's guidance given 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 details will be described by the plant designer for NRC approval at the design certification phase. The staff concludes that this response does not address other issues on testing. ASME/ANSI OM-10, referenced in Section XI of the ASME Code, 1989 Edition, provides for relaxation in the valve testing frequency from quarterly to cold shutdowns or refueling outages if testing during these plant conditions is not practical. However, these rules were written to address testing of valves in currently operating reactors where the detailed piping system designs were completed before promulgation of the IST requirements and do not accommodate quarterly testing. The advanced passive reactors, for which the final design is not complete, have sufficient lead time available to include provisions in their piping system designs to allow testing at power for most valves. Quarterly testing is the base testing frequency in the code and the original intent of the code. Also, it may be necessary to test more frequently than during cold shutdowns or at every refueling outage to provide a component reliability commensurate with the importance of the safety functions to be performed and with system reliability goals. Therefore, to the extent practical, the passive ALWR piping systems should be designed to accommodate the applicable code requirements for quarterly testing of valves (i.e., quarterly full-stroke exercising of check valves - refer to OM-10, Section 4.3.2) rather than allowing designs that only accommodate testing during cold shutdowns or refueling outages. The staff will require vendor applicants to clearly identify those valves that cannot be tested at power and provide justifications for the test frequency of these valves.

Moreover, the passive system designs should incorporate provisions (1) to permit all critical check valves to be tested for performance in both the forward and reverse flow* directions and (2) to verify movement of each check valve's obturator during IST by observing direct instrumentation of the valve such as a position indicator or by nonintrusive test methods.

As previously stated, the staff position on the IST requirements for the essential non-safety-related components will be determined when the staff completes its review of the regulatory treatment of non-safety-related systems. However, the details of the scope of such components will be design specific.

While the passive Requirements Document endorses the staff position that passive plant system designs should permit full-flow testing of installed safety-related check valves to demonstrate operability of the valves under operating conditions, the passive Requirements Document does not address the

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* Check valve performance in the reverse flow direction may be by any of a number of different methods, for example, monitoring an upstream pressure indicator; monitoring tank level; performing a leak test, system hydrostatic or pressure test; radiography; or other nonintrusive technique.

other issues described above (e.g., quarterly full-flow testing of non-safety-related but essential check valves, forward and reverse flow testing, verification of obturator movement). The staff will evaluate the issue of check valve testing methods during its review of individual applications for FDA/DC in accordance with the above positions.

In its letter dated May 17, 1991, the staff requested that EPRI revise Section 12.2.7.3 in Chapter 1 of the passive Requirements Document 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). In the DSER for Chapter 1, the staff stated that testing method and testability are important to reliability assurance and that a commitment to the staff's position on full-flow testing of check valves should be part of the passive Requirements Document.

The staff notes that Revision 4 of the passive Requirements Document requires that "essential" valves be designed and tested in accordance with Section XI of the ASME Code. The systems containing "safety-related" check valves will also be designed to accommodate full-flow testing. Since the passive Requirements Document endorses the staff position that passive plant system designs should permit full-flow testing of installed safety-related check valves to demonstrate operability of the valves under operating conditions, this concern is resolved.

The staff position on the IST requirements for the essential non-safety-related components (e.g., quarterly full-flow testing of non-safety-related but essential check valves) will be determined when the staff completes its review of the regulatory treatment of non-safety-related systems.

In its letter dated May 17, 1991, the staff requested additional information regarding the use of nonintrusive diagnostic techniques for check valves. In Revision 4 of Section 12.2.7.5 of Chapter 1 of the passive Requirements Document, EPRI states that check valve and system design will include provisions for nonintrusive diagnostics use. 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 that EPRI require the plant designer to commit to test hydraulically and pneumatically operated valves in accordance with forthcoming ASME/ANSI OM-18, "Performance Testing of Hydraulic Operated Valve Assemblies in LWR Plants," and OM-19, "Performance Testing of Pneumatically Operated Valve Assemblies in LWR Plants." In its letter dated August 1, 1991, EPRI responded to this request by stating that it was committed to available and applicable codes and standards. When equipment is designed, the plant designer will identify applicable revisions of each document. In the DSER for Chapter 1, the staff stated that the response was not entirely acceptable. The staff's position is that designs should incorporate provisions to test hydraulically and pneumatically operated valves under design-basis differential pressure and flow. The design-basis capability of these types of valves will be expected to be verified before installation, before startup, and periodically through a program similar to that recommended for motor-operated valves in GL 89-10, dated June 28, 1989. Revision 4 of Section 12.2.7.1 Chapter 1 of the passive Requirements Document now requires that all essential valves (including hydraulically and pneumatically operated)

be tested according to the applicable edition of Section XI of the ASME code. The code may not require periodic testing under full design differential pressure and flow. The new Section 12.2.3 of Chapter 1 of Revision 4 to the passive Requirements Document only requires qualification testing of safety-related valves (including hydraulically and pneumatically operated as well as motor-operated valves) to full design differential pressure and flow. EPRI committed to include the following in Revision 4 to the passive Requirement Document: "[p]eriodic testing of installed valves under design differential pressure and flow will be performed in situ where practical." However, Revision 4 does not incorporate this commitment. The staff will verify in each plant-specific application that design-basis capability of safety-related power-operated valves (hydraulically and pneumatically operated valves) is verified before startup and periodically. Pending a commitment, to the staff's positions on preoperational and periodic testing, the staff will evaluate this issue during its review of individual applications for FDA/DC in accordance with the above positions.

In its letter dated May 17, 1991, the staff requested additional information on motor-operated valve (MOV) testing requirements. Revision 4 of Sections 12.2.7.7 of Chapter 1 of the passive Requirements Document 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 its 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 is not required. In the DSER for Chapter 1, the staff concluded that nonactive MOVs in safety-related systems either should be designed to prevent mispositioning or should be required to be subjected to qualification testing to demonstrate capability to recover from mispositioning. The staff stated that 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. Revision 4 to the passive Requirements Document clearly indicates that all-safety related valves (including MOVs) will be subject to qualification testing during which the mispositioning of valves (including nonactive valves) under design-basis fluid conditions will be considered. This is consistent with the staff's regulatory position as stated in GL 89-10 and is, therefore, acceptable.

In the DSER for Chapter 1, the staff identified several MOV design-related concerns that it suggested be addressed in the passive Requirements Document. These issues were identified from recent industry experience with MOVs and as a result of NRC inspections of MOV programs. For example,

- Plant designers should provide MOV operator loads as a function of fluid temperature (subcooling) and seismic/dynamic effects, as well as precise internal dimensions of the valve.

- The valve 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.

- Provisions should be made for the measurement of both stem thrust and actuator torque because of the importance of information regarding the conversion of torque to thrust (i.e., stem factor).

- Containment isolation valves (CIVs) should be designed and test-qualified to be able to isolate flow resulting from a pipe break at the worst-case differential pressure.

Because Revision 4 to the passive Requirements Document adequately provides specific requirements to address each of these concerns, the staff finds them acceptable. Therefore, these concerns are resolved.

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 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 it letter dated August 1, 1991, EPRI referred to Table B.1-1 of Appendix B to Chapter 1 of the passive 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 passive 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 Subsection IWV 3421-3427(a) of Section XI of the ASME Code or Section 4.2.2.3 of ASME/ ANSI OM-10, 1990.

In its letter dated August 1, 1991, EPRI responded to the staff's request by explaining that the requirement in Section 6.2.2.2 of Chapter 5 of the passive Requirements Document is 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 passive Requirements Document, which requires the plant designer to provide for the testing of essential valves in accordance with

ASME/ANSI OM-10. Furthermore, EPRI stated that the designation of specific IST requirements was beyond the scope of the passive Requirements Document and properly belonged in design certification documentation.

The staff approved the use of OM-10, which was incorporated into 10 CFR 50.55a(b), through the 1989 Edition of Section XI, by rulemaking effective September 8, 1992. OM-10 was approved with modifications related to containment isolation valve leakage testing. Specifically,

> When using Subsection IWV in the 1988 Addenda or the 1989 Edition of Section XI, Division 1, of the ASME Boiler and Pressure Vessel Code, leakage rates for Category A containment isolation valves that do not provide a reactor coolant system pressure isolation function must be analyzed in accordance with paragraph 4.2.2.3(e) of Part 10, and corrective actions for these valves must be made in accordance with paragraph 4.2.2.3(f) of Part 10 of ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM-1987.

Nevertheless, OM-10 relaxed somewhat the testing requirements for CIVs by permitting valve combinations rather than specific valves to be analyzed.

Therefore, the staff's position is that all CIVs that are included in the Appendix J program should be included in the IST program as Category A or A/C valves. These CIVs should be tested either individually or in groups as allowed by the code (i.e., OM-10). Where two or more valves on a containment penetration are tested as a group, limiting leakage-rate values must be assigned to the group for the purpose of monitoring valve condition and taking corrective action. If the limiting values are exceeded, the licensee will be required to take actions to determine the leakage path. If plant designers used valve grouping to limit the number of valves that will be subjected to Type C testing, they should design the plant so that leakage of any valve in the group can be identified. The plant designer might also suggest target leakage limits for the group(s) of valves on the diameter of the smallest valve in the group or based on a conservative limit established to another criterion not related to the diameter of the valve. Licensees will be required to comply with the analysis of leakage rates and corrective action requirements of Section XI, Subsections IWV-3426 and IWV-3427(a), or OM-10, Section 4.2.2.3 to the extent practical for group leakage testing.

For the reasons discussed above and because EPRI's position that the specification of IST requirements is beyond the scope of the Utility Requirements Document, the staff will review the issues of analysis of leakage rates and corrective action for CIVs during its review of individual applications for FDA/DC.

12.2.3 Pumps

In its letter dated August 1, 1991, EPRI submitted responses to the staff's May 17, 1991, request for additional information (RAI) concerning the measurement of suction pressure and the full-flow testing of safety-related pumps. In this response, EPRI stated that the passive ALWR will not have safety-related pumps and that the staff's positions in the RAIs should not apply to non-safety-related pumps. As discussed in Section 12.2.1 above, the specific staff position on the IST requirements for the essential non-safety-related

components will be determined in accordance with the process for determining the regulatory treatment of non-safety-related systems, as described in SECY-94-084. Therefore, the staff will evaluate these issues during its reviews of individual applications for FDA/DC.

Section 12.4.1.3 of Chapter 1 of the passive Requirements Document requires that systems be designed so 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 result from 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 is consistent with the guidance given in GL 89-04, "Guidance on Developing Acceptance Inservice Testing Programs," on this issue, and is, therefore, acceptable.

In its letter dated May 17, 1991, the staff requested that EPRI provide a commitment to develop a program that will establish the frequency and extent of disassembly and inspection of safety-related pumps. The staff intends that such a program be an ongoing program that would rely on in-plant and industry data for determining which pumps would require periodic disassembly. The extent of disassembly would also be based on the specific problems identified as a result of this trending.

In its letter dated August 1, 1991, EPRI responded, stating that there are no safety-related pumps in the passive ALWR design. As discussed in Section 12.2.1 above, the specific staff position on the IST requirements for the essential non-safety-related components will be determined in accordance with the process for determining the regulatory treatment of non-safety-related systems, as described in SECY-94-084. Therefore, the staff will evaluate these issues during its reviews of individual applications for FDA/DC.

12.2.4 Conclusions

The staff concludes that, with the exception of the vendor- or utility-specific items noted above, the information in Section 12 of Chapter 1 of the passive Requirements Document satisfies the 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 passive 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. This will be accomplished

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 resin trap cleaning.

Conclusion

The staff concludes that the design features of Section 12 of Chapter 1 of the passive Requirements Document do not conflict with the applicable guidance in RG 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, they are acceptable.

# 13 CONCLUSION

Chapter 1 of the Passive Requirements Document

The staff concludes that the requirements established in Chapter 1 of the passive 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 if they meet the Commission's regulations, guidance, and practices. However, 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 EPRI passive Requirements Document complies with the Commission's regulations. The conclusions regarding the staff's review of the entire passive Requirements Document are provided below. Applicants referencing the passive Requirements Document will be required to demonstrate compliance with the additional guidance in the SRP (NUREG-0800), or provide justification for alternate means of implementing the associated regulatory requirements.

Passive Requirements Document

The staff concludes that the requirements established in the Requirements Document for passive plant designs (Volume III) 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 passive design will be adequate. Therefore, applicants referencing the passive Requirements Document will be required to demonstrate compliance with the additional guidance in the SRP (NUREG-0800), or provide justification for alternative means of implementing the associated regulatory requirements.

In staff requirements memoranda, 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-93-087 and SECY-94-084. When the Commission has completed its review of the basis for the approach that the staff is proposing for those issues, it may at some future point in the review determine that such issues involve policy questions that the Commission may wish to consider. The approaches to resolving these issues have not been reviewed by the Commission and, therefore, 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 Requirements Document for passive plant designs (Volume III) specifies requirements that, if properly translated into a design in accordance with the NRC regulations in force at the time the design is submitted, should result in a nuclear power plant 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 Commission policies on severe-accident protection and public safety goals.

# 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 the passive Requirements Document through Revision 4, submitted by letter dated December 18, 1992. Additional revisions were submitted by letters dated April 5 and 23, May 3, and June 7 and 8, 1993. 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.

Use of the Probabilistic Risk Assessment in Future Designs

It is not the Commission's intent that probabilistic risk assessments (PRAs) for passive plants merely be used to validate an already frozen design, but rather that the PRAs be used as a design tool to optimize the design, enhance safety, and help provide valuable insights into specific plant vulnerabilities. Consistent with this philosophy, the Commission expects passive plant vendors to make use of PRA insights to help ensure 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 passive plant PRAs not focus on checking only the quality of the PRA, but also on how well the vendor or utility continues the use of the PRA to gain insights about key contributors to risk and to minimize the estimated core damage frequency and offsite consequences. Accordingly as indicated in the DSER, the passive Requirements Document should have provided guidance on how to integrate PRAs into the design process and define what documentation should be provided regarding the use of PRAs in this process. Since this guidance was not included in the passive Requirements Document, the staff will review individual applications for FDA/DC to ensure that passive design vendors provide a discussion of how PRA insights were integrated into the design process.

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 design reliability assurance program (D-RAP) as part of the design certification. A plant-specific PRA will provide a technical basis for many aspects of the D-RAP and will need to be integrated into the development and application of the D-RAP.

### 1.1 Review Criteria

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

## 1.2 Scope and Structure of Appendix A to Chapter 1

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

Key topics addressed in the staff's review of Chapter 1, Appendix A, include

- scope of the PRA
- initiating events
- success criteria
- uncertainty analysis
- plant modeling
- human reliability
- containment analysis
- reliability data

The staff has evaluated parts of Appendix A to Chapter 1 that are of particular relevance to passive design features.

## 1.3 Policy Issues

During its review of Appendix A to Chapter 1 of the passive Requirements Document, the staff did not identify issues that involved 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 passive Requirements Document contained the following open issues:

Open Issues

P.1A.0-1 reporting of core damage frequency results as mean values (1.7)

P.1A.0-2 point-estimate quantification (1.8)

P.1A.0-3 quantitative treatment of uncertainties (1.9)

P.1A.0-4 guidance on presenting results of PRA (1.10)

P.1A.0-5 guidance on modeling detail required to represent passive system behavior (2.1)

P.1A.0-6 guidance on modeling interactions between passive and active systems (2.1)

P.1A.0-7 guidance for developing the success criteria for passive systems (2.3)

P.1A.0-8 determination of an appropriate mission time (2.10)

P.1A.0-9 requirements to address important areas of uncertainty specific to the passive plant design (6.1)

P.1A.0-10 failure rate for the main step-up transformers (8.2)

Confirmatory Issues

None

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

1.5 Vendor- or Utility-Specific Items

The DSER for Appendix A to Chapter 1 listed the following vendor- or utility-specific items. The final disposition of each item is discussed in the appropriate section of this appendix, as indicated by the parenthetical notation following each item.

P.1A.V-1 treatment of long-term decay heat removal in the PRA (1.6)

P.1A.V-2 justification of mission items and success criteria (2.10)

P.1A.V-3 reliability data (2.11)

P.1A.V-4 review of core-damage-sequence binning (4.1)

P.1A.V-5 review of actual groupings of the accident sequences into plant damage states (4.2)

P.1A.V-6 review of the evaluation of containment leakage paths (4.3)

P.1A.V-7 computer codes for in-plant sequence assessment (4.5)

P.1A.V-8 verification that the reference site parameters identified in Annex B are consistent with revised 10 CFR Part 100 (5.2)

P.1A.V-9 differences in computer codes used for calculating offsite consequences (5.2)

P.1A.V-10 source terms for representative accident sequences (6.3)

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

P.1A.V-1 integration of PRA insights into the design process (1)

P.1A.V-2 sensitivity analyses and treatment of uncertainties (1.6, 1.9, 5.1, and 6.2)

P.1A.V-3 adequacy of logic models to represent passive plant (2.1)

P.1A.V-4 adequacy of success criteria for passive systems (2.3)

P.1A.V-5 justification for applicability of reliability data (2.11)

P.1A.V-6 bounding analyses of site-specific external events (3)

P.1A.V-7 core-damage-sequence binning (4.1)

P.1A.V-8 containment system analysis (4.2)

P.1A.V-9 containment isolation assumptions (4.3)

P.1A.V-10 containment bypass sequences (4.4)

P.1A.V-11 in-plant sequence assessment (4.5)

P.1A.V-12 containment event analysis (4.6)

P.1A.V-13 use of MAAP code for source term calculations (4.7)

P.1A.V-14 plant release categories (4.8)

P.1A.V-15 adequacy of offsite consequence codes (5.2)

P.1A.V-16 comparison of PRA and physically based source term (6.3)

P.1A.V-17 human reliability analysis (7)

## 1.6 Overall Scope and Methods of Appendix A to Chapter 1

Appendix A to Chapter 1 of the passive Requirements Document defines the ALWR Utility Steering Committee's requirements for the PRA to be performed for passive 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 where existing procedures are incomplete or conflicting. Such a PRA is required to be performed to satisfy the regulatory requirements in 10 CFR Part 52.

### Long-Term Decay Heat Removal

Passive reactor designs differ from current designs in that they will rely on the use of both passive and non-safety active systems to bring the reactor to a cold-shutdown condition. However, Section 1.1 of Appendix A to Chapter 1 does not explicitly address the scope of the PRA analysis required to demonstrate that a stable state can be reliably achieved and maintained in the long term through use of passive and appropriate non-safety features. Although the long-term decay heat removal (DHR) capability of the proposed passive systems offers potential advantages over current active systems, the staff has identified a number of items that must be addressed by the plant designer before it can make a decision regarding the acceptability of this aspect of the passive plant designs (see Section 3.3 of Chapter 5 of this report for a detailed discussion). These include (1) treatment, within the scope of the PRA, of the

plant evolutions necessary to bring the plant to a shutdown condition characterized by a coolant temperature of 420 °F and (2) demonstration of acceptable functional performance and availability of the non-safety defense-in-depth systems to bring the reactor to this shutdown condition and to reach and maintain cold-shutdown conditions.

In response to staff questions in this area, EPRI indicated that the passive Requirements Document would be revised to include (1) an addition to Section 2.3.3.7 of Chapter 1 to require that "all operating conditions (including shutdown ones) shall be taken into account" in the PRA analysis and (2) additional requirements in Appendix A to Chapter 1 to encompass shutdown conditions within the PRA. The Appendix A modifications will require a Level 1 analysis and a simplified evaluation of release frequencies and magnitudes for shutdown conditions. EPRI also proposed an additional section in the passive Requirements Document providing guidance for the conduct of a simplified PRA for conditions other than full power. The staff has concluded that for advanced passive designs, PRA-based evaluation should be conducted to investigate vulnerabilities to external events such as internal floods and fires initiated during modes of operation other than full power. Revisions 3 and 4 of the EPRI passive Requirements Document Section 2.3.3.7 of Chapter 1 and Section 2.12 of Appendix A, included supplemental guidance for a simplified PRA for conditions other than full power.

The EPRI modifications are noteworthy and go a long way toward addressing staff concerns regarding the functional assessment of plant response and the systematic treatment of accidents initiated in modes other than full-power operation. Based on the above considerations and the evaluations described in Chapter 5, the staff concludes that the scope of the PRA for passive plant designs must include system functions (active and passive) necessary to bring the reactor to cold shutdown and to maintain this condition for the long term. The PRA must model (1) all equipment that the NRC requires to be available in passive plants to prevent or mitigate severe accidents and (2) all equipment that an operator is likely to attempt to use to prevent or mitigate any initiating event that may lead to core damage. This should include consideration of accident-management measures or strategies for establishing and maintaining long-term cooling or containment heat removal. Mission times for associated plant equipment must be justified and consistent with this objective as discussed in Section 2.10 of this appendix. The staff generally agrees with the content of Section 2.12.1 ("Identification of Initiating Events") of Appendix A to Chapter 1. However, more emphasis should have been given to investigate the applicability of experience data, and to the consideration of spurious and inadvertent actuation of passive features. The staff will evaluate these aspects of the PRA during its review of an individual application for FDA/DC and will emphasize the use of sensitivity analysis to identify and characterize key human errors and failures controlling risk for these modes of operation.

External Events

The staff has recommended that the Commission approve the following staff position regarding analyses of external events beyond the design basis:

- The staff will require that the analyses submitted in accordance with 10 CFR 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.

- In performing the FDA/DC 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.

The requirements and review approach for external events will be based on the technical positions described in SECY-93-087. Section 3 of this appendix includes further discussion of external events.

1.7 Definition of Core Damage

Section 1.2 of Appendix A to Chapter 1 of the passive Requirements Document states that core damage from a nonreactivity excursion event will be assumed to have occurred if, 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 has been reached in any node of the core as defined in a best-estimate thermal-hydraulic calculation.

For reactivity accidents, core damage is assumed to occur if the axial peak, radial average temperature exceeds 280 cal/g. The value of 280 cal/g is acceptable to the staff as a definition of onset of core damage for reactivity excursion events.

Core Damage Frequency Requirement

In response to the staff's request for additional information (RAI), EPRI revised Section 1.2.3 of Appendix A to read that the plant design will 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). In the DSER for Appendix A to Chapter 1, the staff stated that the wording was still unclear, since there is no precise definition of an "estimate." In addition, point estimates do not permit investigation of the effect that uncertainties may have on the insights from a risk assessment. Although the Commission finds the goal of 1E-5 event per year useful for designers and for potential customers of passive plant designs, it has stated that it does not intend to regulate advanced plants to a specific numerical core damage frequency goal.

In the DSER, the staff stated that 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. (See also Section 1.9 of this appendix regarding uncertainty analyses.)

In the DSER, the staff required that the EPRI guidance be revised to require reporting of mean value core damage frequency results, wherever possible, and identified this as an open issue. In Revision 3, EPRI changed Section 1.2.3 to require that core damage frequencies be reported as mean values. Therefore, this issue is closed.

1.8 Point-Estimate Quantification

Section 1.3 of Appendix A to Chapter 1 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. This section 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.

In the DSER for Appendix A to Chapter 1, the staff stated that, as noted in Sections 1.9 and 6 of this appendix, a point-estimate quantification by itself was not adequate and must be supplemented by an uncertainty analysis with uncertainties propagated from basic event uncertainties, including uncertainties on phenomenological issues. In Revision 5, EPRI revised the requirements of Section 1.3. These changes include the statements that the frequency of core damage must be a mean value and that the magnitude of releases and offsite consequences will be obtained on the basis of the propagation of mean values, when possible, for primary events in Level 2 and Level 3 analyses. The staff finds these changes to be acceptable. The next section includes more detailed discussions of uncertainties and propagation of uncertainties.

1.9 Uncertainty Treatment

Section 1.4 of Appendix A to Chapter 1 originally stated that a qualitative uncertainty analysis would be performed as part of the PRA, and that this analysis would, at 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. It also stated that, where necessary, the qualitative analysis would 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, the passive Requirements Document did not distinguish between the uncertainty and sensitivity analyses that would be performed for the Level 1 versus the Levels 2 and 3 portions of the analysis, or for internally versus externally initiated events.

In response to these staff concerns, EPRI proposed modifications to Section 1.4 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. It stated that a new section (2.5.3) would also be added to Appendix A to require

the analyst to examine the dominant contributors to determine if any logical combinations (sequence cutsets) include events whose failure rates would be correlated.

In the DSER for Appendix A to Chapter 1, the staff stated that the EPRI response represented an improvement of the passive Requirements Document but fell short of adequately addressing the full range of the staff's concern. In particular, the document still did not require a quantitative assessment of uncertainty, or provide guidance regarding 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 recognized 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 (particularly for passive plant designs) and the considerable benefits of performing an uncertainty analysis (particularly for Level 1), the staff considered the guidance on uncertainty analyses in the passive Requirements Document to be deficient. This was identified as an open issue in the DSER.

With regard to Level 2 uncertainties, the staff agrees with the rationale provided by EPRI 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 in the past been labeled "modeling" uncertainties. The staff also agrees with the EPRI statement that it is important that the analysts give careful, systematic consideration to the sources of uncertainty that could be important, and the impact each of these sources might have on the results. Nevertheless, in view of the significant differences between the passive and current plant designs and the general lack of test data, operating experience, and PRA analyses for passive designs, the staff believes that quantitative treatment of the uncertainties in a limited number of Level 2 containment event trees is warranted for passive designs.

In Revision 5, EPRI made further changes to Section 1.4. These changes included general guidance for treatment of uncertainties in both Level 1 and Level 2 segments of the PRA. Furthermore, Revision 4 changes in Section 6 included a statement that a comprehensive set of uncertainty and sensitivity studies is needed to provide adequate perspective to uncertainties in PRA results and the significant contributors to risk. Revision 4 of Section 6.1 included a statement that the propagation (of uncertainties) will be supplemented with well-defined sensitivity studies to investigate sources of uncertainty that do not readily lend themselves to quantification. Additional guidance on treatment of uncertainties in Level 2 analysis was provided in Revision 5, of the passive Requirements Document. These changes represent further improvement of the passive Requirements Document.

The staff believes that the guidance provided in the latest revision of the passive Requirements Document, if followed in spirit by the ALWR vendors, would provide an adequate treatment of uncertainties in the PRA. Therefore, this DSER open issue is closed. The staff will review the applicant's sensitivity analyses and treatment of uncertainties during its review of an individual application for FDA/DC. For the Level 1 portion of the PRA, the staff will require a full uncertainty analysis to be performed with uncertainties propagated from basic events, including initiating event frequencies, data, common cause/mode failure, success criteria, and human error.

Uncertainty analyses will be expected to provide reasonable assurance that the PRA reflects variability in (1) the significance of key actions, events, and phenomena for the plant design and (2) the effectiveness of the accident prevention systems and potential design improvements. The uncertainty analysis for passive designs should also reflect the closer coupling between containment performance and core-melt prevention, and the phenomenological uncertainties inherent in determining passive system success criteria.

With regard to the Level 2 uncertainty analysis, the staff will require applicants for FDA/DC to systematically assess phenomenological issues and challenges of potential risk significance to the ALWR design by means of quantitative sensitivity analysis and explicitly address those issues of risk significance within the Level 2 event trees. Essential elements of this process include (1) initial screening of issues for applicability to the passive design, (2) sensitivity analyses to further delineate issues of potential risk significance, and (3) systematic analysis of issue uncertainty as part of a broader assessment of uncertainty in the overall risk measures. It should be noted that the treatment of uncertainties for Level 2 issues need not be as extensive as that of NUREG-1150, but must be such that the staff has reasonable assurance that the PRA reflects the significance of key actions, events, and phenomena for the plant design and the effectiveness of the accident-mitigation systems.

1.10 Form of Results

In response to staff requests for additional information on Section 1.5 of Appendix A to Chapter 1, EPRI significantly expanded its guidance on presenting the results of the PRA. In the DSER for Appendix A to Chapter 1, the staff stated that the revised section provides the analyst with adequate guidance to ensure that the PRA documentation provides a traceable path and enough information so that a competent PRA analysis team can duplicate the results. However, the staff recommended that in addition to the expanded guidance in Section 1.5, the passive Requirements Document should require reporting of the following:

- the frequency of challenging passive decay heat removal (passive residual heat removal or isolation condenser), and the leading contributions to this frequency (the combinations of failures most likely to cause this event)

- the frequency of challenging the passive inventory makeup systems, and the leading contributions to this frequency

- the frequency of challenging the depressurization function, and the leading contributions to this frequency (for some designs, this will be the same as challenging passive inventory makeup systems; for others, it will not)

- the conditional probability that the depressurization function will fail to reduce the reactor coolant system pressure to the point at which gravity injection can function as designed, and the leading contributions to this event

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- the conditional probability that the staging of the depressurization will fail in such a way as to affect the fuel adversely (e.g., excessive blowdown caused by opening too many valves or the wrong valves first), and the resulting occupational exposure to the workforce

- the frequency with which depressurization will actuate spuriously, and the leading contributions to this event

This was identified as an open issue in the DSER.

In response to the DSER open issue, EPRI supplemented the guidance in Section 1.5 by requiring the above information. These changes were introduced in Revisions 4 and 5. The staff concludes that these changes are acceptable. Therefore, this DSER open issue is closed.

In addition, in the DSER, the staff recommended that the passive Requirements Document provide PRA guidance on providing a thorough assessment of the conditional probability that the sources of gravity injection (core makeup tanks, incontainment refueling water storage tank, gravity drain cooling system, etc.) will, for some reason, be unable to perform their functions as required (e.g., lack of inventory, improper chemistry, human error, leaks).

The staff also identified guidance on presenting PRA results as an open issue in the DSER.

EPRI's response to this issue, contained in a letter dated January 20, 1993, indicated that the sensitivity studies required in Section 6 of Appendix A to Chapter 1 were provided for that purpose. The staff will review applications for FDA/DC to ensure that this information and its underlying assumptions and supporting analyses are provided. Therefore, this DSER open issue is closed.

1.11 Regulatory Treatment of Non-Safety Systems (RTNSS)

In Revision 6, EPRI provided its requirements for the implementation of the regulatory treatment of non-safety systems. Section 1.6 of Appendix A to Chapter 1 has been added, requiring the passive plant designer to take the following steps in order to develop appropriate regulatory oversight for non-safety systems:

- determine safety- and risk-significant non-safety SSC functions and their reliability/availability (R/A) missions relied on to meet NRC requirements, safety goal, and containment performance guidelines using the criteria specified in Section 1.6.2 of Appendix A to Chapter 1 of the passive Requirements Document

- provide a systematic evaluation of adverse safety interactions

- perform a quantitative sensitivity study (focused PRA) to assess the need for non-safety systems to back up passive plant safety systems when challenged

- construct a focused PRA from the comprehensive Level 3 PRA prepared in accordance with Appendix A to Chapter 1 (baseline PRA)

- non-safety SSCs used to prevent the occurrence of initiating events will be subject to regulatory oversight applied, commensurate with their R/A missions for prevention

- construct the focused PRA by removing, as a minimum, the effect of non-safety defense-in-depth functions and their support defined in Section 2.3.1 of Chapter 3, from the baseline PRA event tree logic

- determine what combination (if any) of non-safety SSCs are necessary (in addition to those used to prevent initiating events) to meet NRC regulations, safety goal guidelines (core damage frequency of 1E-4 per year and large release frequency of 1E-6 per year) and the containment performance objectives

- identify and document risk significant non-safety SSC functional R/A missions from the focused PRA which are found to be necessary to meet the safety goal guidelines, containment performance goals, and other NRC requirements

- propose appropriate regulatory oversight measures for the defined R/A missions

The NRC position on the regulatory treatment of non-safety systems in passive ALWRs is described in SECY-94-084 (Annex B of Appendix B of Chapter 1 of this report) and the associated SRM dated June 30, 1994. The staff concludes that the requirements of Section 1.6 of Appendix A to Chapter 1 appropriately implement the RTNSS process insofar as the passive Requirements Document is concerned. However, it will evaluate the acceptability of the implementation of the RTNSS process during its review of an individual application for FDA/DC.

## 2 PLANT MODELING

EPRI must caution PRA analysts to pay particular attention to modeling tanks and their interconnecting or support piping and lines because of the extremely high reliance placed on them and their attendant high importance in preventing core damage from occurring during severe accidents. Examples of potentially important modeling areas in the Westinghouse AP600 design include the coupled (for pressure balancing) pressurizer, cold leg, and core makeup tanks. Another coupling includes the core makeup tanks and the automatic depressurization system (ADS), where each stage of the ADS fires at successively lower levels in the tanks.

### 2.1 Model Structure

Section 2.1 of Appendix A to Chapter 1 of the passive Requirements Documents states that the plant will be modeled in terms of a set of initiating events, event sequences composed of function or system success or failure, 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.

In Revision 5, EPRI supplemented the passive Requirements Document with Section 2.11.3.5 to provide general guidance for evaluating uncertainties for equipment with limited experience data.

The staff notes that the possible ways for functions or equipment to fail are modeled in fault trees. The considerations and techniques needed to model event trees for passive ALWR designs may be different from those for evolutionary ALWR designs. It has not been established that the functional reliability of passive systems can be discussed adequately in terms of the kind of logic models that have been used widely in the analysis of today's plants. The failure probabilities of these systems may be dominated by combinations of circumstances that are poorly represented by true or false variables, especially since small differential pressures are responsible for some of the passive injection functions. In the DSER for Appendix A to Chapter 1, the staff stated that EPRI should provide guidance that states that the language in the passive Requirements Document regarding modeling is the minimum level of sophistication required, and that more careful and detailed modeling of some systems (e.g., consideration of a continuous or more finely discretized spectrum of selected physical process variables) may be necessary to adequately represent system behavior.

Use of a more finely discretized model may be necessary for passive ALWR designers unless a binary-logic scheme can be devised that either provides the desired insights into system behavior or serves as a suitable basis for quantifying functional failure probability. Consider the following hypothetical example in a passive injection system. Partially open valves in one line, in conjunction with back-leakage through a partially seated check valve in

another line, might alter the relatively small differential pressures in a way that substantially reduces the effectiveness of gravity injection. It may be difficult to model such a situation adequately in terms of Boolean variables. If the conservative approximation is made that a small specified amount of check valve blockage is total failure, and a corresponding amount of back-leakage is total failure, and if a very considerable amount of work is done to establish accurate rates for such events, it is doubtful whether the resulting core damage frequency (CDF) will be deemed acceptable. The staff identified this as an open issue in the DSER. The staff will assess the adequacy of the logic models used to represent the passive plant during its review of an individual application for FDA/DC. Therefore, this open issue is closed.

In the DSER, the staff also stated that EPRI should provide guidance on modeling potential systems interactions between passive and active systems, since the philosophy of passive plants appears to be to use non-safety-grade active systems as the first line of defense in the event of a transient. The staff identified this as an open issue in the DSER.

In Revision 5, responding to the open issue, EPRI supplemented Section 2.6.1 by adding a statement requiring a systematic examination for the potential for adverse interactions among systems and in particular for the possibility that any mode of operation of a non-safety system could interfere with a passive safety system in the successful performance of its mission.

The staff considers these additions to Section 2.6.2 and its associated rationale to be satisfactory. Therefore, this open issue is closed. The staff review of design-specific PRAs will include a review of the details of this systematic search process, evaluation of its underlying rationale, and insights drawn from the search results.

## 2.2 Initiating Events

Section 2.2 of Appendix A to Chapter 1 states that the analyst will develop a comprehensive list of potential initiating events for consideration in the PRA, including events while the plant is shut down. EPRI indicates that the analyst should consider input from summaries of operating experience for current-generation plants, consider PRAs for plants with similar design characteristics, and review system designs. In response to the staff's RAI dated September 23, 1991, EPRI revised Section 2.2 to state that analysts will review each passive 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 to mitigate the consequences. In addition, the staff recommends that Section 2.2 be revised as follows:

- It should indicate that there may be site-specific or plant-specific initiating events (e.g., loss of service water) whose accurate quantification will depend on the specific plant design.

- The list of initiators should be expanded to include often neglected low-frequency events leading to core damage that may significantly challenge prevention and mitigation equipment, and low-frequency events with very high consequences.

- The list of initiators should be expanded to include shutdown events such as spurious actuation of passive systems and failures of the reactor shutdown cooling system.

The search for initiating events must be conducted in such a way that events that are capable of affecting plant safety can be identified even at frequencies on the order of a few percent of the core damage frequency. Since it is expected that passive 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 by analysts or were considered to have been bounded by "traditional" initiators. Three examples illustrate initiators not normally considered in PRAs, but that may be important for passive plants: (1) multiple initiators (e.g., a loss of service water followed by an independent loss of offsite power before service water has been recovered), (2) low-frequency loss-of-coolant accidents (LOCAs) (e.g., multiple steam generator tube ruptures), and (3) multiple operator errors of commission.

## 2.3 Success Criteria

Section 2.3 of Appendix A to Chapter 1 requires that a definition of success and failure for each function be provided that is 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 conservatism 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.

In the DSER for Appendix A to Chapter 1, the staff stated that it agreed that the measures discussed in the passive Requirement Document were prudent and believed the basis for these measures given by EPRI was valid. However, it considered the guidance in the passive Requirements Document to be inadequate in that it failed to (1) identify development of success criteria for passive designs as fundamentally different from, and potentially more complex than, that for current plant designs and (2) provide further guidance in this regard.

With nearly exclusive reliance on active, pumped systems, the development of success criteria for current plant designs is relatively straightforward and typically involves identifying the set of active components (e.g., pumps) or trains that must properly operate in order to satisfy a given function. System performance is treated in a bimodal fashion (i.e., the system is either operable or failed) with degraded performance of a component or function generally considered to be a failure. Furthermore, the effectiveness of the systems in performing the required function is ensured, given that they operate.

In contrast to current plant designs, the performance and effectiveness of passive plant features is a continuous rather than a discrete function. Development of success criteria for a passive plant design is further complicated by such factors as (1) inability to easily characterize the effectiveness of the passive mechanisms (e.g., thermal-hydraulic uncertainties, the flow rates and heat-removal effectiveness for natural circulation when

initiated late in an accident or during shutdown), (2) reliance on both active and passive systems to perform certain functions, and (3) potential interactions between active and passive mechanisms (e.g., injection by an active system at the wrong time may cause stagnation of natural circulation flow). A relatively large number of detailed thermal-hydraulic analyses for combinations of active and passive system failures and successes are necessary to establish the success criteria for a single phase of an accident. These criteria are likely to be different for other (e.g., later) phases of the accident, for other modes of operation (e.g., events initiated at cold shutdown), and for situations in which the sequence timing or characteristics have been significantly altered by operator actions (e.g., operators attempt to use (but fail) active systems to prevent the automatic initiation of passive safety systems).

Success criteria will have to be extended to modes of operation not normally considered in a PRA (i.e., modes other than full power and hot shutdown). In particular, front-end success criteria will no longer stop at core cooling or containment cooling, but will need to address both what is required to bring the plant to cold shutdown and what is needed to bring the plant to a static shutdown condition of 420 °F that can be maintained without the supply of any consumables (e.g., makeup water) and without operator action.

As a result, development of success criteria may become much more demanding and the most important part of developing the PRA. The staff believes that the fundamental differences between active and passive design concepts will necessitate a different approach to defining the success criteria, and that the passive Requirements Document should provide additional guidance on this aspect of the PRA. This was identified as an open issue in the DSER.

The staff has further evaluated this subject in Section 1.9 of this document. A possible approach to address the uncertainties in passive features success criteria is to characterize the sources of uncertainties in the performance of this feature and use this information to select ranges for sensitivity parameters and develop a reasonable strategy for the sensitivity analyses and use the results of sensitivity analyses to define the margin available beyond assumed success criteria. The staff will assess the adequacy of success criteria during its review of an individual application for FDA/DC. Therefore, this open issue is closed.

## 2.4 Sequence Logical Identity

Section 2.4 of Appendix A to Chapter 1 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. 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 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 purposes of 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 response to the staff's RAI dated September 23, 1991, the guidance has been modified 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 has been revised to state explicitly that truncation will be performed after recovery actions (which are cutset dependent in many cases) are accounted for. This is acceptable to the staff.

### Nested Solution Process

Section 2.5.2 of Appendix A to Chapter 1 discusses the nested solution process, which is apparently 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 response to the staff's RAI dated September 23, 1991, EPRI modified the guidance acceptably 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. However, the method of truncation of the accident sequences must be documented carefully in the PRA, so that the method can be evaluated when the staff reviews the PRA.

## 2.6 Modeling of Dependencies

Section 2.6 of Appendix A to Chapter 1 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 the 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. EPRI also recognizes that consideration of the potential for common-cause failure of groups of components in different systems may be relatively more important in passive plants because of fewer

intersystem dependencies (i.e., fewer support systems). The staff agrees that the proper modeling of dependencies is essential when conducting any PRA. It finds EPRI's guidance in this area acceptable.

2.7 Interaction and Modeling of the Containment Systems

Section 2.7 of Appendix A to Chapter 1 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. Attention to this aspect of the PRA is particularly important for passive plant designs, since the performance of containment and reactor systems is more closely coupled for these designs.

2.8 Common-Cause Failures

Section 2.8 of Appendix A to Chapter 1 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. 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 specifies that the EPRI systematic human action reliability procedure (SHARP) analysis framework will be used for considering human interactions in the PRA because an approach is needed that shows consistency, traceability, and realism. EPRI requires that the analysis deal explicitly with the following:

- definition of human actions
- screening for importance
- task breakdown
- representation in relation to systems logic models
- iteration between human and hardware modeling
- quantification
- documentation

The SHARP analysis requires a disposition of each of the following types of human interactions:

- interactions before an initiating event (Type A)
- actions related to the initiating event (Type B)
- actions following an initiating event dictated by procedures (Type CP)
- actions following an initiating event to recover equipment or systems to terminate a sequence (Type CR)

Each of these types of human interactions requires a different approach in the PRA.

The staff's evaluation of the EPRI PRA key assumptions and groundrules for human reliability analysis is given in Section 7 of this appendix.

2.10 Mission Time

Section 2.10 of Appendix A to Chapter 1 originally stated that a mission time of 24 hours will be used for equipment required to remain operable for successful core cooling after an initiating event. Results of recent PRA studies performed by the French indicate that long-term (up to 1 year) reliability of post-LOCA core cooling using recirculation from the containment sump can appreciably contribute to core-melt frequency. On the basis of their findings that post-LOCA decay heat removal through operation of the emergency core cooling system 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 passive plant designs in absolute terms, the results of the French studies suggest that greater attention should be paid in the PRA to events outside the 24-hour time window that has been traditionally assumed in the PRA. This may be particularly important for passive plant designs in that they use a combination of passive and non-safety active systems to perform long-term residual heat removal, and the reliability of these systems may be less than for current plant designs. In this DSER for Appendix A to Chapter 1, the staff stated that in view of the potential risk contribution from plant operations later than 24 hours, it considered the guidance on mission time in the passive Requirements Document to be inadequate.

The staff concluded that (1) the PRAs performed for passive plant designs should include treatment of the plant evolutions and system functions (active and passive) necessary to bring the reactor to cold shutdown and a safe shutdown condition, and to maintain either of these conditions for the long term, and (2) mission times should be determined and justified by the plant designer consistent with this expanded scope. This will necessitate consideration of mission times considerably longer than 24 hours for certain equipment and functions, as well as explicit treatment of actions by onsite operating staff and offsite support organizations that may need to be accomplished within this timeframe. Determination of an appropriate mission time should be based on consideration of (1) passive and active system performance and reliability late in an event, (2) availability and reliability of available backup systems or components, and (3) actions required to be taken by operating staff and offsite response organizations, and the provisions that would be in place to ensure such actions could be taken in a timely manner. Mission times may be different for going to cold shutdown and static shutdown. The different types of equipment and operator actions needed to go to these modes must be reflected explicitly in the system and function success criteria (see Section 2.3). This was identified as an open issue in the DSER.

In response to the staff concerns raised in the DSER, EPRI has supplemented Section 2.10 in Revision 4 of Appendix A to Chapter 1 to require that (1) actions that must be taken beyond 24 hours be considered explicitly and (2) sensitivity of the results to the selection of mission times be evaluated

for systems that provide long-term core cooling and containment heat removal. Frequencies of core damage and of serious release will be evaluated in two sensitivity studies, one in which relevant equipment will be required to continue operating for 72 hours, and another in which the mission time is at least 30 days. The staff concludes that these changes are acceptable and significantly improve the guidance in the passive Requirements Document. Therefore, this DSER open issue is closed. In any case, mission times and success criteria will have to be justified by the plant designer and will be reviewed by the staff as part of the FDA/DC process for each passive design PRA.

2.11 Reliability Data

Section 2.11 of Appendix A to Chapter 1 discusses reliability data and suggests the use of a database given in Annex A of Appendix A. In response to the staff's RAI dated September 23, 1991, EPRI modified Annex A of Appendix A to Chapter 1 to indicate that the failure data used in any PRA will have to be reviewed for their applicability to each particular design, considering the operational environment of the equipment and the procedures applicable to the equipment. These remarks apply to all the reliability data, including common-cause-failure parameters.

In Revision 5, EPRI also supplemented the passive Requirements Document with Section 2.11.3.5, which recognizes that data may be available only for comparable equipment used in different ways in existing (non-passive) plants. EPRI also stated that probability distributions for the failure rates and other reliability data should reflect any increased uncertainty resulting from the use of any equipment for which experience in applications similar to those planned for passive plants is limited. The staff interpretation of these statements is that the new distributions for the failure rates will not only reflect increased uncertainties but also adjustment of their mean values as a result of differences in application. With this understanding, the changes in the passive Requirements Document are acceptable to the staff.

In the DSER, the staff concluded that test and maintenance intervals should be considered when estimating the reliability of equipment. Some of the most important passive features are nearly untestable in that full tests can or will never be performed of the as-built systems (e.g., the automatic depressurization system in the AP600 design) because, for example, their testing would empty primary coolant into the containment. In some cases, safety-significant equipment will have long test intervals. Any safety-significant motor-operated valve (MOV) that has a long test interval (MOV failures are well represented by a standby-stress model) would have a very 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 passive safety systems if additional random failures or human errors should occur.

In the DSER, the staff also concluded that reliability data for non-safety-related (but normally safety-related 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. This was identified as an open issue.

During its review of an individual application for FDA/DC, the staff will review the applicant's justification regarding applicability of EPRI's reliability data to their design. Furthermore, at the combined license stage, the staff will review the site-specific loss-of-offsite-power frequency and the site-specific probability of recovery of offsite power as a function of duration of its loss. Therefore, this DSER open issue is closed.

## 3 EXTERNAL EVENTS

Lessons learned from past PRA studies indicate that internal floods, fire, and seismic events can be important contributors to risk. Analysis of seismic and fire events is complex, and the magnitude of their contribution to risk is associated with substantial uncertainties. This has led to the following staff positions, which were included in SECY-93-087.

- The staff requires that the analysis submitted in accordance with 10 CFR 52.4 include an assessment of internal and external events. PRA will be used to support margins-type assessment of seismic events. A PRA-based seismic margins analysis will consider plant-level and sequence-level high confidence, low probability of failures (HCLPFs) and fragilities for all sequences leading to core damage or containment failures up to approximately twice the magnitude of the safe shutdown earthquake (SSE). Simplified probabilistic methods, such as EPRI's Fire Induced Vulnerability Evaluation (FIVE) methodology, will be used to evaluate fires. Traditional probabilistic techniques should be used to evaluate internal floods.

- The staff also requires that ALWR vendors perform bounding analyses of site-specific external events likely to be a challenge to the plant such as river flooding, storm surge, tsunami, volcanism, high winds, and hurricanes. When a site is chosen, its characteristics should be compared to those assumed in the bounding analyses to ensure that the site is enveloped. If the site is enveloped, the COL applicant need not perform further PRA evaluations for these external events. The COL applicant should perform site-specific PRA evaluations to address any site-specific hazards for which a bounding analysis was not performed or that are not enveloped by the bounding analyses to ensure that no vulnerabilities due to siting exist.

Seismic

The major difference between a seismic PRA and the proposed PRA-based margins approach is that the latter does not convolute fragility curves with hazard curves. Rather, the PRA-based margins approach measures the robustness of the plant to withstand earthquakes of a given g-level. This method eliminates the need to deal with uncertainty in the seismic hazard curve for the site and identifies potential design-specific seismic vulnerabilities. Understanding these vulnerabilities will be useful in developing the reliability assurance program, identifying operator training requirements, and focusing on accident management capabilities.

The staff believes that a plant designer can best determine the seismic capability of the plant through a combined approach that takes advantage of the strengths of both PRA and margins methods. This approach (based on an internal events PRA, its existing event and fault trees, and its random failures and human errors) allows for a comprehensive and integrated treatment of the plant's response to an earthquake. This approach should yield meaningful measures of a proposed design's seismic capability.

The plant designer should use plant logic models covering the various systems that could lead to core damage. Typically, this would be accomplished by modifying design-specific 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 then determine all important accident sequences using the event trees and fault trees (based on fragility data for each component for each sequence). In addition, the designer would determine the value of the minimum HCLPF for the plant by determining the HCLPF values for the important structures, systems, components for each accident sequences. The HCLPF values calculated in this manner can be used to measure the plant's robustness and to provide an acceptable estimate of the earthquake ground motion that the plant is expected to be able to survive without core damage. In general, the value of the plant HCLPF should be at least twice the design ground motion zero period acceleration. If not, the designer should perform a more detailed evaluation to identify any vulnerability against which the plant requires strengthened protection. HCLPF calculations also indicate which components and systems limit the seismic capability of the plant.

The EPRI seismic margins methodology (EPRI NP-6041) limits the search for success paths to two, and will not provide some important insights concerning human error and random failure. Therefore, this methodology does not appear adequate for evaluating evolutionary and passive ALWR designs. In addition, the EPRI margins method is limited in its useful application for passive plant margins analyses as is the NRC margins method used in the individual plant examination of externally initiated events, since these margins approaches are based on insights from PRAs of operating plants (PWRs and BWRs). The staff concludes that it would be inappropriate to rely exclusively on either of these methods to perform seismic margins analyses for passive designs. Instead, the staff recommends the use of a PRA-based margin-type assessment, as discussed below.

As discussed in Section II.N of SECY-93-087, which is provided as Annex A of Appendix B to Chapter 1 of this report, the staff believes that a well-designed passive plant would inherently result in a plant HCLPF of approximately twice the magnitude of its SSE. The staff is concerned that if a PRA-based margins analysis only looks at accelerations significantly below twice the magnitude of the SSE, the analysis could effectively screen out potential design-specific seismic vulnerabilities. The staff would anticipate that analysis of a fully developed seismic PRA will identify seismic events as significant contributors to risk (perhaps the largest, given the significant estimated reduction in internal event core damage frequency in advanced ALWRs). Therefore, it is important to fully understand potentially significant seismic vulnerabilities and other seismic insights. This information would be captured by a PRA-based seismic margins analysis that considers sequence-level HCLPFs and fragilities for all sequences leading to core damage or containment failures up to approximately twice the magnitude of the SSE.

In its staff requirements memorandum (SRM) dated July 21, 1993, concerning the technical position in Section II.N of SECY-93-087, which is provided as Annex A of Appendix B to Chapter 1 of this report, the Commission approved the use of PRA-based seismic margins assessment methodology. Furthermore, in the SRM, the Commission provided additional guidance and directed that a PRA-based seismic margins analysis consider sequence-level HCLPFs and fragilities for

all sequences leading to core damage or containment failures up to approximately 1.67 times the ground motion acceleration of the Design Basis SSE.

Fire and Internal Floods

Fire events can also be evaluated using simplified methods such as EPRI's FIVE methodology rather than a comprehensive PRA study. Using PRA-based simplified methods such as FIVE will help the designer focus on design robustness and its capability to withstand the consequences of fire events.

For internal floods, the staff concludes that the designer should use traditional probabilistic techniques similar to those used in recent PRA studies. In its letter of August 21, 1992, EPRI provided comments regarding the staff's proposed requirements for analysis of certain internal and external events. EPRI concluded that its requirements for separating redundant safe-shutdown equipment by physical barriers designed to withstand full-height compartment flooding should make core-damage events resulting from internal flooding very unlikely.

The staff concludes that separated safety divisions do not necessarily preclude internal floods from being significant contributors to risk. The staff notes that evolutionary ALWR designs that include physical separation between safety divisions appear to respond better to internal fires and floods than do traditional designs. However, although this physical separation should reduce the expected frequency and severity of these events, it does not preclude them from being important contributors to risk. For example, an internal flooding PRA study for an evolutionary design has already identified several design improvements that are needed to achieve the desired level of internal flood protection. The staff believes that the systematic evaluation required during the development of an internal flooding PRA is necessary to ensure that potential flood vulnerabilities are identified.

Site-Specific Hazards

Details of the specific site characteristics will likely not be available until the combined license review stage. The staff intends to require COL applicants to perform a site-specific PRA evaluation that addresses all applicable site-specific hazards (such as river flooding or upstream dam break, storm surges, tsunamis, volcanism, hurricanes, or nearby transportation or explosion hazards). The staff will review these site-specific PRA evaluations to ensure that no vulnerabilities are introduced by siting the standardized plant at a location where external hazards could pose an unacceptable or unanticipated risk.

However, the staff will require that ALWR vendors perform bounding analyses of site-specific external events likely to be a challenge to a plant. When a site is chosen, its particular siting characteristics can then be compared to those of the bounding analyses in order to minimize the potential for the site-specific PRA evaluations to identify significant site-specific weaknesses for the standard design. Before FDA/DC, the staff will evaluate fires, internal floods, and other external events that are not site dependent and will evaluate submitted bounding analyses for site-specific external events.

The COL applicant should submit the results of this comparison and of the site-specific PRA evaluations to the NRC at the COL review stage. For vendors of passive ALWR designs who have submitted an application for design approval to the NRC, the staff will require that these bounding analyses be submitted and reviewed by the staff before the FDA/DC is issued. For subsequent design approval of evolutionary and passive ALWRs, the staff will require that the applicant submit bounding analyses for external events with the application.

At the COL stage, the staff will review the site-specific characteristics to ensure that bounding analyses at the design stage properly envelope the site-specific events. The staff plans to conduct walkdown inspections to confirm that design commitments have been met.

# 4 CONTAINMENT ANALYSIS

Section 4 of Appendix A to Chapter 1 of the passive Requirements Document discusses the containment analysis. The staff's conclusions on how the guidance should be revised are given below. However, the staff will still have to review the containment analysis portion of a PRA submitted for a passive design 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, 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 (1) a means of managing the number of accident sequences and (2) 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. As EPRI states, it is necessary that all sequences within a bin lead to similar effects with respect to containment and source term phenomena. The staff will review core-damage-sequence binning when the design-specific PRA is submitted.

## 4.2 Containment System Analysis

Section 4.2 of Appendix A to Chapter 1 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 review the actual groupings of the accident sequences into plant damage states when the design-specific PRA is submitted.

## 4.3 Containment Isolation

Section 4.3 of Appendix A to Chapter 1 requires that containment penetrations be accounted for in the evaluation of containment leakage paths and that they 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 valves have been mispositioned)

- 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 review the actual implementation of these assumptions when the design-specific PRA is submitted.

## 4.4 Containment Bypass

Section 4.4 of Appendix A to Chapter 1 originally required that containment bypass sequences be assessed and include all connections to the reactor coolant system. In Revision 4, EPRI modified Section 4.4 to state that "only connections which may lead to significant releases will be taken into account." EPRI states that containment bypass sequences can result in significant releases from the containment and have the potential to be important risk contributors. Examples of 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)

The staff agrees that containment bypass sequences are important and, therefore, agrees with the EPRI requirement to assess such sequences. The treatment of containment bypass sequences will be evaluated by the staff during its review of an individual application for FDA/DC.

## 4.5 In-Plant Sequence Assessment

Section 4.5.2 of Appendix A to Chapter 1 states that the MAAP code will be the primary tool used to assess the thermal-hydraulic and other physical processes involved in 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 MAAP and fails to establish a requirement for dealing with the 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. A particular concern for passive plants is that the adequacy of the code for predicting natural circulation flows has not been demonstrated by EPRI and assessed by the staff.

In its response to a staff RAI, EPRI proposed a modification to the passive Requirements Document, which is incorporated in Revision 3, to further clarify, by example, the situations in which use of codes other than MAAP might be more appropriate 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 adequately addresses the staff's concern regarding the adequacy of the MAAP code for determining success criteria. The staff intends to review applications 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.

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 issue is discussed in Section 4.6 of Appendix A to Chapter 1 of the passive Requirements Document and Sections 1.9 and 4.6 of this appendix.

## 4.6 Containment Event Analysis

Section 4.6.1 of Appendix A to Chapter 1 states that the containment event tree (CET) will comprise the important phenomenological issues associated with containment loading and/or source term evolution, and as the supporting rationale indicates, the CET provides an excellent means to identify and quantify important phenomena. The section further lists elements that should be considered for the ALWR in the development of the CET.

The staff considers that the identified list of phenomena provides a reasonable characterization of the types of issues that should be addressed by a plant designer, but cautions that this set of phenomena may not necessarily be complete, particularly with regard to passive plant designs. Passive plant designs differ from current designs in that containment and reactor system performance are more closely coupled. For example, in passive designs, loss of containment integrity and associated coolant inventory from the containment may seriously compromise the passive core cooling function, whereas such a breach of integrity might be tolerated in a conventional design if active system (pump) operability could be maintained. Furthermore, because of the lack of familiarity and experience with the passive designs, potentially important containment performance issues may not be obvious at this time. The staff concludes that although the list of phenomena in this section is reasonable, it will review individual applications for FDA/DC to ensure that plant designers have appropriately screened for other risk-significant containment challenges. The staff's review of EPRI's requirements to resolve the larger issue related to Level 2 uncertainty analysis is provided in Section 1.9 of this appendix.

Section 4.6.2 of Appendix A to Chapter 1 states that potentially important phenomena that are not currently addressed in the MAAP code will be addressed in the containment event analysis and identifies two such phenomena in the rationale portion of this section (steam explosions and hydrogen deflagration due to equipment or operator failures). The staff agrees with this requirement, but notes that the two phenomena identified are by no means the complete set; for example, core debris coolability should also be addressed because MAAP models for this phenomenon are highly questionable. More importantly, the EPRI requirement should not apply only to phenomena that are not explicitly treated using MAAP. Rather, PRA analysis should consider all potentially risk-significant phenomena for possible inclusion in the containment event analysis. This matter is also discussed by the staff in Section 1.9 of this appendix.

Section 4.6.3 of Appendix A to Chapter 1 states that the CETs will be quantified using best-estimate point values. The staff considers the guidance in this section to be insufficient, given that risk estimates may be dominated by uncertainties in modeling and assumptions related to severe-accident progression and phenomenology.

In its response to staff concerns, EPRI proposed modifications to the passive Requirements Document that would 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 the MAAP code. The rationale 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 of one approach is "to develop a probability distribution for an event, reflecting 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 modifications to the passive Requirements Document adequately address most of its concerns regarding Level 2 uncertainty analysis, but fall short in that they 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 passive Requirements Document also fails to provide a requirement and guidance for screening issues to identify which might require this more extensive treatment. This is discussed in greater detail in Section 1.9 of this appendix, as part of the open issue regarding uncertainty treatment.

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

should 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 to be gained from a Level 2 analysis can be obtained using 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.

## 4.7 Source Term Definition

Section 4.7 of Appendix A to Chapter 1 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. (Resolution of the staff's concerns regarding the use of the MAAP code is discussed in Section 4.5 of this appendix.) Accordingly, the staff will address this issue during its review of an individual application for FDA/DC.

## 4.8 Plant Release Categories

Section 4.8 of Appendix A to Chapter 1 states that similar end points of the containment analysis may be grouped into release categories for use in the in-plant consequence analysis. EPRI states that past PRAs have shown that containment event tree 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, and types and amounts of isotope fractions released.

The staff agrees that containment event tree 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.

# 5 OFFSITE CONSEQUENCES

Section 5 of Appendix A to Chapter 1 of the passive Requirements Document discusses the assumptions for the offsite consequence analysis portion of the PRA.

## 5.1 Implementation of the Public-Safety Requirement

Section 5.1 of Appendix A to Chapter 1 requires that a complementary cumulative distribution function (CCDF) be developed for whole-body dose at a distance of 0.5 mile from the reactor. 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 risk requirement if an individual located at a distance of 0.5 mile has a probability of less than 1E-6 per year of experiencing a whole-body dose of 25 rem or more. While the CCDF is a well-accepted method of visually displaying risk curves, a point-estimate CCDF provides an incomplete picture of risk and is inadequate. The staff concludes that uncertainties must be propagated through the analysis, and a family of CCDF curves developed, each with their degree of belief (in addition to the mean CCDF curve). The staff will evaluate this information as part of the FDA/DC review of each passive ALWR PRA. The staff's review of EPRI's requirements regarding the treatment of uncertainties is discussed in Section 1.9 of this appendix.

The staff notes that while the goal of 1E-5 per year is useful for designers and for potential customers of passive 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" should be reported to the extent possible.

## 5.2 Method for Offsite Consequence Analysis

Section 5.2.1 of Appendix A to Chapter 1 states that a reference site with the characteristics listed in Annex B of Appendix A will be used for calculating offsite consequences for the ALWR. Annex B contains meteorology 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.

As part of design certification for each passive ALWR, the staff will require ALWR vendors 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 design alternatives (SAMDAs) 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 calculating cancer risk. In the DSER for Appendix A to Chapter 1, the staff noted that Version 1.5 of MACCS used the results of the BEIR III study* 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 risk coefficients for fatal cancers would be approximately doubled or tripled by the model modifications (memorandum dated April 11, 1990, from J. M. Taylor, NRC, to Chairman K. M. Carr, "Evaluation of Recent Reports on Health Effects of Low-Level Ionizing Radiation"). In the DSER, the staff concluded that until these modifications were incorporated into MACCS, the use of CRAC2 was acceptable. The staff subsequently released a modified version of the MACCS code that incorporates the results of the BIER V study. Accordingly, the staff will require the use of the MACCS code for calculating offsite consequences for passive ALWR designs. The staff will address the adequacy of offsite consequences codes during its review of an individual application for FDA/DC.

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*National Research Council, Committee on the Biological Effects of Ionizing Radiation, "The Effect on Populations of Exposure to Low Levels of Ionizing Radiation (BEIR III)," National Academy Press, Washington, D.C., 1980.

**National Research Council, Committee on the Biological Effects of Ionizing Radiation, "Health Effects of Exposure to Low Levels of Ionizing Radiation (BEIR V)," National Academy Press, Washington, D.C., 1990.

# 6 UNCERTAINTY AND SENSITIVITY ANALYSES

Section 6 of Appendix A to Chapter 1 of the passive Requirements Document 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. 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 regard to uncertainty in the PRA results and the significance of potential contributors to risk. In the DSER for Appendix A to Chapter 1, the staff did not agree with a number of the positions taken by EPRI. Subsequently, EPRI provided additional guidance, which is addressed below. In addition, in Revision 6 of the passive Requirements Document, EPRI added specific guidelines for performance of the quantitative sensitivity study (focused PRA) required by the process developed by industry and the NRC to determine the RTNSS in passive plants. The staff has evaluated EPRI's requirements for implementation of the RTNSS process in Section 1.11 of this appendix.

## 6.1 Analysis of Systems and Sequences

Section 6.1 of Appendix A to Chapter 1 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, or that are unique to passive plant designs
- failure rates for components for which data are very limited
- common-cause parameters, especially those whose assessment relied heavily on engineering judgment to compensate for a lack of experimental data
- contribution of non-safety or non-passive systems in preventing core damage
- human interactions following an initiating event
- seismic hazard

EPRI states that these are the primary areas that may be most uncertain or are likely to be important to the frequency of core damage. The staff agrees that these are important areas of uncertainty that must be addressed explicitly in the passive Requirements Document.

In the DSER, the staff accepted the scope defined for sensitivity analysis. It recognized that the guidance did not lack details about the manner and range of variation for the parameters selected for sensitivity studies analysis, but stated that it did not agree that sensitivity studies alone were an adequate substitute for uncertainty analysis. 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 also indicated in the DSER the need for conducting importance analysis in support of reliability assurance programs. Among the importance measures recommended by the staff are the risk achievement worth, the risk reduction worth, and equivalent measures of importance such as Fussel-Vesely and Birnbaum importance measures.

In the DSER, the staff required EPRI to provide guidance on how a full uncertainty analysis will 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 error. 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 and (2) the effectiveness of the accident-prevention systems and potential design improvements. The staff also noted in the DSER that the Level 1 uncertainty analysis for passive designs will be more complex than that for current plants because of the closer coupling between containment performance and core-melt prevention, and because of the phenomenological uncertainties inherent in determining passive system success criteria. These deficiencies were identified as an open issue in the DSER.

The staff has reviewed the changes to Section 1.4 of Appendix A to Chapter 1 made in Revision 5, concerning treatment of uncertainty, and the changes to Section 6 made in Revision 4. The staff concludes that the new requirements represent a significant improvement in the passive Requirements Document guidance on treatment of uncertainty. Therefore, this open issue is closed. The staff's review of EPRI's requirements regarding treatment of uncertainty is discussed further in Section 1.9 of this appendix.

6.2 Assessment of Containment Response

Section 6.2 of Appendix A to Chapter 1 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. The staff agrees that uncertainties in these parameters are

important but notes that this list is not necessarily complete. For example, the potential for suppression pool bypass should be assessed in the analysis of pressure suppression containment designs.

In the DSER for Appendix A to Chapter 1, the staff noted that sensitivity analyses alone might not be sufficient to address certain containment issues or challenges. In particular, because of the close coupling between reactor and containment system performance and the general lack of familiarity or experience with passive plant designs (see Section 4.6 of this appendix), some Level 2 issues might be appropriate for inclusion in the full uncertainty analysis to be performed for Level 1. This is especially true for those issues or parameters that are critical to defining system success criteria (see Section 2.3).

Responding to this concern in Revision 5, EPRI modified Section 6.2 to provide further clarification regarding the scope and objectives of required sensitivity analyses and the treatment of uncertainties with regard to important severe-accident phenomena and challenges. The staff concludes that the guidance now provided in the passive Requirements Document, if followed in spirit by the ALWR vendors, would provide for an adequate consideration of uncertainties in the Level 2 PRA. The staff will review the applicant's sensitivity analysis and treatment of uncertainties during its review of an individual application for FDA/DC.

## 6.3 Analysis of Source Terms

Section 6.3 of Appendix A to Chapter 1 lists the following parameters that could affect the magnitude of accidental radionuclide releases that must be investigated in appropriate sensitivity studies: (1) the effectiveness of containment scrubbing mechanisms and (2) 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. In the DSER for Appendix A to Chapter 1, the staff agreed that it was appropriate to consider these parameters in sensitivity studies, but again stated that sensitivity studies alone might be insufficient. As stated in Section 1.9 of this appendix, the staff will require ALWR vendors to systematically identify risk-significant issues of relevance to the passive plant designs and to reflect the uncertainties in those issues within the Level 2 analysis. Although the thrust of this systematic process is on severe-accident phenomena and challenges, it is also important that plant designers identify and assess those issues that may result in a shift of accident frequencies from relatively benign 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. This would include consideration of uncertainties in source term issues.

In the DSER, the staff noted that EPRI had provided a new requirement to Section 4.10 of Appendix A to Chapter 1, which requires that designers confirm that the PRA source terms for representative accident sequences for their design are bounded by the physically based source term used in licensing

calculations. EPRI has submitted additional changes to Section 2.4.1.2 of Chapter 5 to provide guidance on how such a comparison would be made.

The passive Requirements Document now requires the plant designer to compare the fission product release to the environment for functional sequences with frequency greater than 1E-7 per year against the releases to the environment from the updated design-basis source term. Cases for which the PRA estimated release exceeds the design-basis source term release shall be reported and explained. This comparison is intended to confirm that there are no probabilistically important sequences with fission product release greater than the design-basis accident.

The staff considers this acceptable and will evaluate this analysis as part of the design certification review for each ALWR PRA.

# 7 HUMAN RELIABILITY ANALYSIS

The staff notes the similarity in the Requirements Document guidance for evolutionary (Volume II) and passive (Volume III) designs. For this reason, the staff evaluation of the human reliability guidance in Section 2.9 of the passive Requirements Document is identical to that provided in Section 7 of the FSER for Chapter 1, Appendix A, for Volume II.

The staff concludes that the guidelines provided by EPRI for developing and maintaining documentation on human reliability analysis (HRA) and for reporting the results are acceptable. The staff will review the applicant's HRA treatment (with a focus on novel features in passive designs that may require special treatment of the role of the human element) during its review of an individual application for FDA/DC.

## 8 CONCLUSION

On the basis of its review and subject to the reservations stated in this report, the staff concludes that the EPRI requirements established in Appendix A to Chapter 1 of the passive 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 if the plant-specific PRAs will be adequate.

Therefore, the staff concludes that Appendix A to Chapter 1 specifies requirements that, if properly translated into a design in accordance with the NRC regulations in force at the time the design is submitted, should result in a PRA 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.

The staff also notes that, because of the general nature of the guidance in the passive Requirements Document, it has accepted the spirit and intentions of the guidance, but it will perform detailed reviews of the vendor submittals in these areas to ensure adequate treatment. Thus, although the EPRI guidance may be judged adequate, many technical judgments regarding the acceptability of approaches for dealing with these issues have, in effect, been deferred until the staff conducts its review of an individual application for FDA/DC.

# ANNEX A - RELIABILITY DATABASE FOR ALWR PROBABILISTIC RISK ASSESSMENTS

Annex A of Appendix A to Chapter 1 of the passive 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 to obtain initiating-event frequencies for loss-of-coolant accidents and for most transient events. Section A2 describes the treatment of the frequency and recovery of losses of offsite power in greater detail. Section A3 summarizes the sources 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 passive ALWRs.

The staff will review the applicant's reliability database and justification of the applicability of the data used during its review of an individual application for FDA/DC.

# ANNEX B - ALWR REFERENCE SITE

Annex B of Appendix A to Chapter 1 of the passive Requirements Document gives the values of parameters expected to conservatively represent the consequences of most potential sites. The staff addressed Annex B in its final SER on the Requirements Document for evolutionary plant designs.

# ANNEX C - COMPONENT SEISMIC CAPACITIES

Annex C, which provided data on the seismic capacity of selected components, has been deleted from Appendix A to Chapter 1 of the passive Requirements Document.

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 of the passive Requirements Document through Revision 5. Revision 5 of the passive Requirements Document was submitted to the NRC by letter dated October 1, 1993. Appendix B to Chapter 1 was prepared, under the project direction of EPRI and the Advanced Light Water Reactor (ALWR) Utility Steering Committee, by Grove Engineering, Inc.; MPR Associates, Inc.; S. Levy Incorporated; TENERA, L.P.; and EPRI.

## 1.1 Review Criteria

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

## 1.2 Scope and Structure of Appendix B

Appendix B to Chapter 1 of the passive Requirements Document is a compilation of those regulatory requirements and guidance that EPRI believes are applicable to the design of passive ALWRs. It gives EPRI's justifications for deviations from current criteria and its proposed resolution of unresolved and generic safety issues that are considered applicable to the passive ALWR design.

## 1.3 Regulatory Requirements and Guidance

Section 1 of Appendix B to Chapter 1 lists NRC regulations and guidance that EPRI believes are currently applicable to LWR design and identifies EPRI's position with respect to each. EPRI states that Section 1 of Appendix B will represent 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 passive Requirements Document when the document is completed. Tables B.1-3 and B.1-4 identify those items that EPRI believes are not applicable to the ALWR design.

Section 10.2 of Chapter 1 of the passive Requirements Document originally stated that the ALWR would comply with the NRC regulatory requirements and guidance in effect as of January 1, 1990, consistent with the commitments in Section 1 of Appendix B to Chapter 1 of the passive 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 (RGs), the Standard Review Plan (SRP), 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 as of 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 passive Requirements Document may be developed as the staff completes its reviews of the detailed design information provided in the FDA/DC applications, and as these designs are litigated in the FDA/DC hearings. Therefore, the staff concludes that the ALWR plant designers will comply with issue resolutions adopted by the NRC staff during its reviews of the FDA/DC applications in accordance with the requirements of 10 CFR Part 52. This will be addressed by the staff during its review of an application for FDA/DC.

Section 10.2.4 of Chapter 1 of the passive Requirements Document 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 compliance commitment specified in Appendix B as follows:

Comply: The comply designation indicates that the ALWR design should fully comply with all regulatory requirements and guidance provided by the reference.

Comply With Intent: The comply with intent designation indicates that the ALWR design should comply with the intent of the regulatory items where the passive ALWR plant takes a fundamentally different design approach to satisfy specific regulatory requirements and guidance items. Clarification of each designated item is provided in Section 4 of Appendix B to Chapter 1 of the passive Requirements Document.

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 passive Requirements Document.

The staff has reviewed Tables B.1-1 through Table B.1-4 of Appendix B and concludes that, except as discussed throughout this report, they are generally acceptable for consideration of the passive plant designs. The staff's review of the entire passive Requirements Document provides a clearer picture of the extent to which the EPRI document complies with the Commission's regulations.

## 1.4 Outstanding Issues

The DSER for Appendix B to Chapter 1 of the passive Requirements Document contained the following outstanding issues:

Open Issues

P.1B.0-1 impact of the elimination of the operating-basis earthquake (OBE) from the design process (2.1.1, 3.3.1, and Item I.M of Annex A)

P.1B.0-2 applicability of industry codes and standards (2.1.1 and Item II.A of Annex A)

P.1B.0-3 tornado design basis (2.1.2 and Item II.F of Annex A)

P.1B.0-4 simplification of emergency planning requirements (2.1.3 and Item III.G of Annex A)

P.1B.0-5 need for leakage control system for main steam isolation valves (2.3.1 and Item II.E of Annex A)

P.1B.0-6 main steam isolation valve leakage rate (2.3.1 and Item II.E of Annex A)

P.1B.0-7 simplification of postaccident sampling system (2.3.2 and Item II.I of Annex A)

P.1B.0-8 containment leak rate testing (2.5.1 and Item II.H of Annex A)

P.1B.0-9 source term (2.5.2 and Item I.A of Annex A)

P.1B.0-10 hydrogen generation and control (2.5.3, 3.2.26, and Item I.G of Annex A)

P.1B.0-11 safe shutdown requirements (2.5.6 and Item III.D of Annex A)

P.1B.0-12 revised deficiency reporting requirements (Issue II.J.4.1) (3.2.3)

P.1B.0-13 criteria for safety-related operator actions (Generic Safety Issue B-17) (3.2.4)

P.1B.0-14 diesel generator reliability (Generic Safety Issue B-56) (3.2.6)

P.1B.0-15 allowable emergency core cooling system equipment outage periods (Generic Safety Issue B-61) (3.2.7)

P.1B.0-16 reactor coolant pump seal failures (Generic Safety Issue 23) (3.2.10)

P.1B.0-17 bolting degradation or failure (Generic Letter 91-17) (Generic Safety Issue 29) (3.2.12)

P.1B.0-18 use of EPRI NP-5076 on good bolting practices for bolted joints (3.2.12)

P.1B.0-19 bolting degradation or failure (Generic Safety Issue 29) (3.2.12)

P.1B.0-20 effects of fire protection system actuation on safety-related equipment (Generic Safety Issue 57) (3.2.13)

P.1B.0-21 power-operated relief valve and block valve reliability (Generic Safety Issue 70) (3.2.14)

P.1B.0-22 anticipated transients without scram (ATWS) (Generic Safety Issue 75) (3.2.16 and Item I.B of Annex A)

P.1B.0-23 unanalyzed reactor vessel thermal stress during natural convection cooldown (Generic Safety Issue 79) (3.2.17)

P.1B.0-24 control room habitability (3.2.18)

P.1B.0-25 low-temperature overpressure protection (Generic Safety Issue 94) (3.2.21).

P.1B.0-26 piping and use of highly combustible gases in vital areas (Generic Safety Issue 106) (3.2.23)

P.1B.0-27 dynamic qualification and testing of large-bore hydraulic snubbers) (Generic Safety Issue 113) (3.2.24)

P.1B.0-28 reliability/operability and on-line testability of the final actuation contacts in protection systems (3.2.25)

P.1B.0-29 essential service water pump failures at multiplant sites (Generic Safety Issue 130) (3.2.28)

P.1B.0-30 guidelines for upgrading other procedures (Generic Safety Issue HF 4.4) (3.2.33)

P.1B.0-31 electronic display of procedures, use of mixed types of procedures from one control station to another, and use of active simulator to validate procedures (3.2.33)

P.1B.0-32 clarification of definition of local control stations (3.2.34)

P.1B.0-33 centralization of safety functions (3.2.34)

P.1B.0-34 development of a human factors verification and validation test plan (3.2.35)

P.1B.0-35 documentation of test activities for traceability and ensuring that all human factors requirements are addressed during test and evaluation (3.2.35)

P.1B.0-36 development of quantitative measures to assess human-system performance (3.2.35)

P.1B.0-37 uniform damping values (3.3.1)

P.1B.0-38 modal combination of high-frequency modes for vibratory loads (3.3.1)

P.1B.0-39 safety implications of control systems (Generic Safety Issue A-47) (3.3.3)

P.1B.0-40 heat exchanger testing (3.3.4)

P.1B.0-41 control of biofouling (3.3.4)

P.1B.0-42 zebra mussel fouling (3.3.4)

P.1B.0-43 mid-loop operation (Item I.C of Annex A)

P.1B.0-44 station blackout (Item I.D of Annex A)

P.1B.0-45 fire protection (Item I.E of Annex A)

P.1B.0-46 intersystem loss-of-coolant accident (Item I.F of Annex A)

P.1B.0-47 core-concrete interaction - ability to cool core debris (Item I.H of Annex A)

P.1B.0-48 high-pressure core melt ejection (Item I.I of Annex A)

P.1B.0-49 containment performance (Item I.J of Annex A)

P.1B.0-50 dedicated containment vent penetration (Item I.K of Annex A)

P.1B.0-51 equipment survivability (Item I.L of Annex A)

P.1B.0-52 inservice testing of pumps and valves (Item I.N of Annex A)

P.1B.0-53 electrical distribution (Item II.B of Annex A)

P.1B.0-54 seismic hazard curves (Item II.C of Annex A)

P.1B.0-55 leak before break (Item II.D of Annex A)

P.1B.0-56 classification of main steamline of BWR (Item II.E of Annex A)

P.1B.0-57 containment bypass (Item II.G of Annex A)

P.1B.0-58 level of detail (Item II.J of Annex A)

P.1B.0-59 prototyping (Item II.K of Annex A)

P.1B.0-60 inspections, tests, analyses, and acceptance criteria (Item II.L of Annex A)

P.1B.0-61 reliability assurance program (RAP) (Item II.M of Annex A)

P.1B.0-62 site-specific probabilistic risk assessments (Item II.N of Annex A)

P.1B.0-63 severe-accident mitigation design alternatives (Item II.O of Annex A)

P.1B.0-64 generic rulemaking related to FDA/DC (Item II.P of Annex A)

P.1B.O-65 regulatory treatment of non-safety systems (252 and Item III.A of Annex A)

P.1B.O-66 definition of passive failure (Item III.B of Annex A)

P.1B.O-67 thermal-hydraulic stability of the simplified BWR (Item III.C of Annex A)

P.1B.O-68 control room habitability (Item III.E of Annex A)

P.1B.O-69 radionuclide attenuation (Item III.F of Annex A)

Confirmatory Issues

P.1B.C-1 commitments to 10 CFR Part 21 and 10 CFR 50.55(e) (3.2.3)

P.1B.C-2 Target Rock safety/relief valves (3.2.5)

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

## 1.5 Vendor- or Utility-Specific Items

The DSER for Appendix B to Chapter 1 contained the following vendor- or utility-specific items. The final disposition of each item is discussed in the appropriate section of this appendix, as indicated by parenthetical notation following each item.

P.1B.V-1 compliance of FDA/DC applications with Commission's regulations and guidance (1.2)

P.1B.V-2 issue resolution for FDA/DC reviews (1.2)

P.1B.V-3 dynamic seismic analysis of main steam piping and condenser (2.3.1 and Item II.E of Annex A)

P.1B.V-4 main steamline piping classification (2.3.1 and Item II.E of Annex A)

P.1B.V-5 seismic analysis of turbine building (2.3.1 and Item II.E of Annex A)

P.1B.V-6 plateout considerations for main steam piping and valves (2.3.1 and Item III.F of Annex A)

P.1B.V-7 reactor vessel level instrumentation system (2.4.1)

P.1B.V-8 fission product leakage control (2.5.2 and Item III.F of Annex A)

P.1B.V-9 fission product cleanup system (2.5.2 and Item III.F of Annex A)

P.1B.V-10 dedicated containment vent penetration (2.5.4 and Item I.K of Annex A)

P.1B.V-11 automatic emergency core cooling system switch to recirculation (Generic Safety Issue 24) (3.2.11)

P.1B.V-12 bolting degradation or failure (Generic Safety Issue 29) (3.2.12)

P.1B.V-13 detached thermal sleeves (Generic Safety Issue 73) (3.2.15)

P.1B.V-14 failure of high-pressure coolant injection steamline without isolation (Generic Safety Issue 87) (3.2.20)

P.1B.V-15 adequate leak test requirements for pressure isolation valves in inservice testing programs and technical specifications (TS) (3.2.22)

P.1B.V-16 identification and design of interfacing systems (3.2.22)

P.1B.V-17 implementation of procedures for electrical power reliability (3.2.27)

P.1B.V-18 leakage through electrical isolators in instrumentation circuits (Generic Safety Issue 142) (3.2.29)

P.1B.V-19 availability of chilled water systems and room cooling (Generic Safety Issue 143) (3.2.30)

P.1B.V-20 reliability of recirculation pump trip during an ATWS (Generic Safety Issue 151) (3.2.31)

P.1B.V-21 loss of emergency service water system (Generic Safety Issue 153) (3.2.32)

P.1B.V-22 spectral peak shifting (3.3.1)

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

P.1B.V-1 applicability of industry codes and standards (2.1.1 and Item II.A of Annex A)

P.1B.V-2 acceptability of passive autocatalytic recombiner system for combustible gas control (2.5.3 and Item I.G of Annex A)

P.1B.V-3 dedicated containment vent penetration (2.5.4 and Item I.K of Annex A)

P.1B.V-4 revised deficiency reporting requirements (Issue II.J.4.1) (3.2.3)

P.1B.V-5 diesel generator reliability (Generic Safety Issue B-56) (3.2.6)

P.1B.V-6 effects of fire protection system actuation on safety-related equipment (Generic Safety Issue 57) (3.2.13)

P.1B.V-7 anticipated ATWS (Generic Safety Issue 75) (3.2.16 and Item I.B of Annex A)

P.1B.V-8 control room habitability (Generic Safety Issue 83) (3.2.18 and Item III.E of Annex A and Item D of Annex B)

P.1B.V-9 failure of high-pressure coolant injection steamline without isolation (Generic Safety Issue 87) (3.2.20)

P.1B.V-10 low-temperature overpressure protection (Generic Safety Issue 94) (3.2.21)

P.1B.V-11 adequate leak test requirements for pressure isolation valves in inservice testing programs and technical specifications (3.2.22)

P.1B.V-12 identification and design of interfacing systems (3.2.22)

P.1B.V-13 piping and use of highly combustible gases in vital areas (Generic Safety Issue 106) (3.2.23)

P.1B.V-14 dynamic qualification testing of large-bore hydraulic snubbers (Generic Safety Issue 113) (3.2.24)

P.1B.V-15 implementation of procedures for electrical power reliability (3.2.27)

P.1B.V-16 essential service water pump failures at multiplant sites (Generic Safety Issue 130) (3.2.28)

P.1B.V-17 leakage through electrical isolators in instrumentation circuits (Generic Safety Issue 142) (3.2.29)

P.1B.V-18 availability of chilled water systems and room cooling (Generic Safety Issue 143) (3.2.30)

P.1B.V-19 loss of essential service water in LWRs (Generic Safety Issue 153) (3.2.32)

P.1B.V-20 decoupling OBE from safe-shutdown earthquake (3.3.1)

P.1B.V-21 spectral peak shifting (3.3.1)

P.1B.V-22 heat exchanger testing (3.3.4)

P.1B.V-23 control of biofouling in raw water systems (3.3.4)

## 2 PLANT OPTIMIZATION SUBJECTS

Plant 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.

The following subsections of the staff's evaluation of EPRI's proposed plant optimization subjects are not numbered sequentially. The staff's evaluation maintains EPRI's convention, which designates subsection accordingly to the Requirements Document chapter containing specific requirements for a particular issue.

### 2.1 Issues Addressed in Chapter 1 of Passive Requirements Document

The plant optimization subjects for the passive plant designs are identified in Section 2 of Appendix B to Chapter 1 of the passive Requirements Document. The staff's evaluation is given in this section under the corresponding section numbers of the passive Requirements Document. The regulatory departure analysis in Annexes A and B of this appendix provides additional information on these subjects.

#### 2.1.1 Operating-Basis Earthquake and Dynamic Analysis Methods

As a plant optimization subject, Section 2.1.1 of Appendix B to Chapter 1 of the passive Requirements Document, EPRI proposes to eliminate the operating-basis earthquake (OBE) as a design-basis event and to use alternative criteria for dynamic analysis methods. In the DSER for Appendix B to Chapter 1, the staff reported that it agreed in principle with the deletion of the OBE from plant design. However, the implementation of this optimization subject, as presented in Appendix B, was not completely acceptable. This was identified as an open issue to be resolved using the staff's recommendations in Policy Issue I.M, "Elimination of Operating-Basis Earthquake." On April 2, 1993, the staff issued SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and ALWR Designs," which included the following recommendations on this policy issue:

(1) To account for earthquake cycles in the fatigue analyses of piping systems performed until new guidance is issued, the staff proposes using two safe shutdown earthquake (SSE) events with 10 maximum stress cycles per event (20 full cycles of the maximum SSE stress range). Alternatively, the staff proposes that the number of fractional vibratory cycles equivalent to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Appendix D of IEEE 344-1987, "IEEE Recommended Practice for Qualification of Class 1E Equipment for Nuclear Power Generating Stations."

(2) The effects of anchor displacements in the piping caused by an SSE should be considered with the ASME Service Level D limit.

(3) When the OBE is eliminated from design, no replacement earthquake loading should be used to establish the postulated pipe rupture and leakage crack locations.

(4) The mechanistic pipe break and high-energy leakage crack locations determined by the piping stress (without the OBE) and fatigue locations may be used for equipment environmental qualification and compartment pressurization purposes.

(5) Equipment should be qualified with five one-half SSE events followed by one full SSE event. Alternatively, a number of fractional peak cycles equivalent to the maximum peak cycles for five one-half SSE events may be used in accordance with Appendix D of IEEE 344-1987 when followed by one full SSE.

The information in Section 2.1.1.2 of Appendix B to Chapter 1 of the passive Requirements Document suggests revisions to 10 CFR Part 100, Appendix A, 13 different RGs, and 12 different sections of the SRP. It allows the plant designer to use all of these changes in the design of the ALWR. These suggested changes do not completely agree with the above staff positions in SECY-93-087. In addition, some of these changes, which involve various requirements in Section 4 of Chapter 1 of the passive Requirements Document, are not acceptable. These issues are discussed in Section 4 of Chapter 1 of this report. Therefore, the staff does not find Section 2.1.1 of Appendix B to Chapter 1 of the passive Requirements Document to be acceptable. The staff will review individual applications for FDA/DC in accordance with the above positions from SECY-93-087. Therefore, this open issued is closed.

2.1.2 Tornado Design

In 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 passive Requirements Document, EPRI originally proposed a maximum tornado design-basis wind speed of 260 mph, based on American National Standards Institute/American Nuclear Society (ANSI/ANS) 2.3, "Standard for Estimating Tornado and Extreme Wind Characteristics at Nuclear Power Sites." The use of ANSI/ANS 2.3 was an exception to RG 1.76, "Design Basis Tornado for Nuclear Power Plants." This was identified as an open issue in the DSER. In Revision 4 to Section 2.1.2 of Appendix B to Chapter 1 of the passive Requirements Document, EPRI states that the ALWR Program would use 300 mph as the maximum tornado design-basis wind speed and refers to revised Table 1.2-6, "Envelope of ALWR Plant Site Design Parameters," in Chapter 1 of the passive Requirements Document. The staff finds that the maximum tornado design-basis wind speed of 300 mph is acceptable. The staff's evaluation and its acceptance of this issue are provided in detail in Section 4.5.2.5 of Chapter 1 and under Item II.F of Annex A to this appendix. Therefore, this open issue is closed.

Elimination of Missile Provisions

Section 2.1.2.3 of Appendix B to Chapter 1 of the passive Requirements Document states that EPRI will eliminate certain missile provisions, such as labyrinths in heating, ventilating, and air conditioning inlets and exhausts, to lower construction costs and enhance maintenance.

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 NRC 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 locating independent redundant systems in compartments located in a missile-protected structure, or by subgrade location at a sufficient depth. The staff concludes that the EPRI-proposed elimination of certain missile provisions may be acceptable, if sufficient justification is provided on a case-by-case basis for compliance with RG 1.117 and SRP Section 3.5.2. However, EPRI's generic justification of cost reduction and maintenance convenience is not a sufficient basis for eliminating missile-protection measures.

2.1.3 Simplification of Offsite Emergency Planning

In Section 2.1.3 of Appendix B to Chapter 1 of the passive Requirements Document, EPRI proposes that the emergency plan regulations for the passive plant be changed to significantly simplify offsite emergency planning. EPRI indicates that this simplification would eliminate the need for certain plume exposure pathway-related requirements, including early notification of the public, analysis of the time required to evacuate and to take other protective actions for various sectors and distances, and provisions for exercising the offsite plan. EPRI states that the onsite emergency plan and a limited number of plume exposure pathway and ingestion pathway planning elements would be retained to provide a base of expansion of response efforts in the event such expansion is necessary. EPRI believes that its design requirements, which it believes provide a low probability of core damage and, in the event of a core damage accident, high assurance of containment integrity and low offsite dose, justify the proposed simplification. The staff discusses this issue under Item III.G of Annex A of this appendix.

EPRI has advised the staff that it is not pursuing the simplification of emergency preparedness issue at this time. In its letter dated December 31, 1993, EPRI stated that it would be providing its preliminary position to NUMARC, who will have the lead for coordinating industry review, deliberation and subsequent action, as deemed appropriate, for the policy aspects.

As discussed in Section 2.6 of Chapter 1 of this report, the staff's position regarding the non-policy portions of emergency planning for passive ALWR designs is discussed in Annex D of this appendix.

2.3.1 BWR Main Steamline Isolation Valves and Leakage Control/Classification of Main Steamline of Boiling-Water Reactor

In Section 2.3.1 of Appendix B to Chapter 1 of the passive Requirements Document, EPRI proposes to eliminate the BWR main steam isolation valve (MSIV) leakage control system and provide an alternate 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 3.3.1 of Appendix B to Chapter 1, Section 3.4.1.5 of Chapter 2, Sections 5.3.3 and 5.4.1.5 of Chapter 3, Section 1.2.3 of Chapter 5, and Section 3 of Chapter 13 of the passive Requirements Document. In Section 6.4.3.7 of Chapter 5 of the passive Requirements Document, EPRI also proposes a maximum allowable MSIV leak rate of 35 cubic feet per minute (cfm) per steamline. In its letters dated May 17 and August 29, 1991, the staff stated its positions relative to 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 were the result of the proposed elimination of the MSIV 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 and leaktight integrity during and following an SSE. Although this issue involves information in Chapters 1, 2, 3, 5, 6, and 13, for continuity, the staff discusses below the basis for its positions and gives its evaluation of all of EPRI's responses.

Main Steamline Classification

Background

The main steamlines in BWR plants, including the SBWR, contain dual quick-closing 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 (TS) limit for MSIV leakage is typically 11.5 standard cubic feet per hour (scfh) per valve. Operating experience has indicated 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, RG 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.

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 (November 1988) and NEDO-31858P (February 1991), both titled "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 leakage control system requirement. The LCS may not be effective for MSIV leakage rates greatly in excess of TS limits because of limitations in its design. Generic Issue C-8, "Main Steamline Valve Leakage Control Systems," was established to address these concerns for currently operating plants.

The following evaluation provides the staff's position on the EPRI-proposed resolution for both evolutionary and passive BWR designs. 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 and leak-tight 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 TS 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 TS 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 resolution approach developed by the BWROG and EPRI 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 are used to mitigate the radiological consequences of an accident that could result in potential offsite exposures comparable to the dose reference values specified 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 10 CFR Part 100 be designed to remain functional during and after a 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 classified as safety-related and seismic Category I. In addition, 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 occurs involve the use of either a suitable dynamic analysis or a suitable qualification test.

Non-Nuclear Experience: In Appendix D to NEDC-31858P, BWROG submitted a report to the staff titled "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 earthquake experience 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 in 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) that any such failure would be contrary to the large body of historical earthquake experience data.

After reviewing the EQE report for condensers, the staff concludes that, in general, the applicability of the experience database to BWR condensers in U.S. plants is appropriate. 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 BWR: Because of the lack of evidence provided by the EQE report to support generic acceptability of the capability 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.

In the DSER for Appendix B to Chapter 1, the classification of the BWR main steamline was identified as an open issue and a vendor-specific issue. Both of these issues are related to Policy Issue II.E, "Classification of Main Steamlines in Boiling-Water Reactors." In Revision 3 to the passive Requirements Document, EPRI responded to these issues by revising Section 3.4.1.5 of Chapter 2. This revision provides requirements that are consistent with all of the staff positions discussed above relative to the impact of the elimination of the BWR MSIV leakage control system on the classification of the BWR main steamlines. On the basis of the above discussion, the staff concludes that the quality and safety requirements for the BWR main steamline from the outermost isolation valve up to, but not including, the turbine stop valve (including branch lines to the first normally closed valve) are equivalent to the guidelines in SRP Section 3.2.2 and RG 1.29 and are acceptable. Therefore, the part of Policy Issue II.E related to the classification of the main steamline is satisfied and the open and vendor-specific issues are closed.

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 passive 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.

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 and leaktightness under SSE loading conditions.

In the DSER for Appendix B to Chapter 1, the staff reported that EPRI's proposal to use the seismic experience database in lieu of a seismic analysis for the line from the first valve of the MSIV leakage pathway to the condenser was not consistent with the staff's positions discussed above and was, therefore, not acceptable. Although this portion of the MSIV leakage pathway does not have to be seismic Category I, the experience database does not provide adequate assurance of seismic integrity for such piping. This issue was identified as part of an open issue related to Policy Issue II.E. In Revision 5, EPRI revised Sections 3.4.1.5 of Chapter 2 and Section 3.3.1.1.2.11 of Chapter 13 of the passive Requirements Document to respond to this issue. Revised Section 3.3.1.1.2.11 of Chapter 13 references revised Section 3.4.1.5 of Chapter 2 for a description of the seismic analysis of the turbine bypass line from the first valve to the condenser. Section 3.4.1.5 of Chapter 2 now contains a requirement that the mathematical model for the dynamic seismic analyses of the main steamlines and branch line piping include the main steam drain lines and turbine bypass lines from the first normally closed valve up to and including the condenser nozzle. This requirement is consistent with the staff position discussed above relative to the classification of the MSIV leakage pathway and is acceptable. Therefore, the part of Policy Issue II.E regarding the classification of piping associated with the main steamline is satisfied and this open issue is closed.

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 structures involved in this issue are given in Section 3.1.1 of Chapter 13 of this report.

Condenser Design: In the DSER for Appendix B to Chapter 1, the staff reported that because the passive Requirements Document did not address the condenser design details and no ALWR plant had been constructed, the staff could not evaluate the applicability of the generic seismic experience database for the ALWR condenser. Thus, as a minimum the staff required that the condenser anchorage be seismically analyzed to ensure that it is capable of maintaining its structural and pressure 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 concluded that the additional safety-related and seismic Category I requirements (e.g., the quality assurance requirement in 10 CFR Part 50, Appendix B, or the inservice inspection requirements in the ASME Code) were 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. This issue was included as part of Policy Issue II.E in Annex A of this appendix. In Revision 5, EPRI responded by revising Sections 3.4.1.5 and 4.4.3.20 in Chapter 2 of the passive Requirements Document. These revisions provide requirements that state that the mathematical model for the dynamic seismic analyses of the main steamlines and branch line piping incorporate the bypass lines up to and including the condenser nozzle. The condenser nozzle is assumed to be the anchor point for the seismic analysis. The anchorages of the condenser will be evaluated for

(1) SSE loads derived from an equivalent static analysis of the condenser and (2) the nozzle loads derived from the bypass line seismic analysis. If the condenser is mounted on spring supports, the additional amplification introduced by this support configuration will be considered. The staff concludes that these requirements provide an acceptable basis to demonstrate that the condenser anchorage will be capable of maintaining its structural integrity during and after an SSE. This part of Policy Issue II.E is resolved.

Turbine Building: As discussed above, the elimination of the BWR MSIV leakage control system results in staff positions that certain non-seismic Category I piping systems and equipment inside the non-seismic turbine building have to be seismically analyzed to withstand the SSE. Position C.2 in RG 1.29 provides guidelines intended to protect such piping and equipment from the possible failure of non-seismic structures such as the turbine building. In the DSER for Appendix B to Chapter 1, the staff identified this issue as one to be resolved during FDA/DC. The staff's evaluation of this issue is given in Section 4.4 of Chapter 6 of this report.

Plant Walkdowns: The staff requires that plant-specific walkdowns of non-seismically designed systems, structures, and components overhead, adjacent to, and attached to the alternative leakage path (i.e., the main steam piping, drain lines, bypass lines, and main condenser) be conducted by the COL applicant before commercial operation to assess potential failures. The walkdowns, which should identify potential failure modes, will provide high confidence that the alternative leakage paths in BWRs can retain their structural integrity during and following an SSE.

Conclusions

The above-described approach for ALWRs to resolve the BWR main steamline issue provides reasonable assurance that the main steam piping from the outermost isolation valve up to and including 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 and leaktight, 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 to be performed during the COL phase on a case-by-case basis. This issue is also discussed under Item II.E of Annex A of this appendix and in Section 3.1.1 of Chapter 13 of this report.

Plateout

The main steamlines in BWRs contain dual quick-closing 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 of the valves will leak. Operating experience has indicated that the leaktightness of MSIVs has occasionally degraded and the specified low leakage has not always been maintained.

When calculating offsite consequences of potential accidents, the staff conservatively assumed that radioactive material passing through the MSIV at

its TS leakage limit is released directly into the environment. No credit is taken for the integrity and leaktightness of the main steam piping and condenser to provide holdup and plateout of fission products.

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 Plant," installation of a supplemental leakage control system (LCS) to ensure that the isolation function of the MSIVs is in accord with the specified limits. Most of the operating BWRs have an LCS.

In response to the MSIV leakage concerns, the BWROG commissioned a program of studies to determine the causes of high leak rates and the means to eliminate them. The results of these studies were submitted to the staff in General Electric (GE) proprietary reports, NEDO-31643P (November 1988) and NEDO-31858P (February 1991), both titled "Increasing Main Steam Isolation Valve Leakage Rate Limits and Elimination of Leakage Control Systems." EPRI referenced these reports in the passive Requirements Document as the basis for not requiring an LCS and specifying a substantially higher MSIV leakage rate (35 scfh) than the typically allowed limit for operating BWRS.

Following a LOCA, three potential release pathways exist for main steam leakage thought the MSIVs:

(1) main steam drain lines to the condenser with delayed release to the environment through the low-pressure turbine seals

(2) turbine bypass lines to the condenser with delayed release to the environment through the low-pressure turbine seals

(3) main steamline (MSL) turbine stop and control valves through the high-pressure turbine seals to the environment bypassing the condenser

The consequences of leakage from pathways 1 and 2 will be essentially the same because the condenser can be used to process MSIV leakage. The condenser iodine removal efficiency will vary depending on the inlet location for the bypass or drain line piping, but in either case, iodine will be removed. For pathway 3, MSIV leakage through the closed turbine stop and control valves will not be processed via the condenser. For this case, the high-pressure turbine (having a large internal surface area associated with the turbine blades) will remove iodine.

The staff believes that as long as either a turbine bypass or a drain line leakage pathway is available, MSIV leakage through the closed turbine stop and control valves (pathway 3) will be negligible. Essentially all of the releases will be through the main condenser because there will be no differential pressure in the MSL downstream of the MSIVs following the closure of the valves.

Furthermore, MSIV leakage through pathway 3, if any, will have been subjected to the same iodine-removal processes in the MSLs (up to turbine stop valves) as the other pathways. The leakage will be further subjected to iodine removal by deposition in the high-pressure turbine internal surfaces. Removal by the main condenser is not applicable in pathway 3.

Basic principles of chemistry and physics predict that gaseous iodine and airborne iodine particulate material will deposit on surfaces. Several laboratory and in-plant studies have demonstrated that gaseous iodine will be deposited by chemical adsorption, and particulate iodine will be deposited through a combination of sedimentation, molecular diffusion, turbulent diffusion, and impaction. The transport of gaseous iodine in elemental and particulate forms also has been studied for many years, and several groups have proposed different models to describe the observed phenomena.

To verify GE's findings in its proprietary reports and to evaluate EPRI's proposal, the staff has developed its own model that treats the MSIV leakage pathway as a sequence of small segments for which instantaneous and homogeneous mixing is conservatively assumed; the mixing computed for each segment is passed along as input to the next segment. The number of segments depends on the parameters of the line and flow rate and can be as many as 100,000 for a long, large-diameter pipe and a low flow. Each line segment is divided into five compartments that represent the concentrations of the three airborne iodine species, the surface that contains iodine available for resuspension, and surface iodine that has reacted and is fixed on the surface.

The staff's model considers three iodine species: elemental, particulate, and organic. A fourth species, hypoiodous acid, was considered for the purpose of the staff's model to be a form of elemental iodine. All iodine in the segment undergoes radioactive decay. The resulting concentration from each segment of the deposition compartment serves as the input to the next segment. The staff's model and technical references indicated that particulate and elemental iodine would be expected to deposit on surfaces with rates of deposition varying with temperature, pressure, gas composition, surface material, and particulate size. Therefore, the staff believes that an appropriate credit for the removal of iodine in the MSLs and main condensers should be provided in the radiological consequence assessment following a design-basis accident.

Sections III(c) and VI of Appendix A to 10 CFR Part 100 require 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 guidelines of Part 100 be designed to remain functional during and after an SSE. Thus, the MSL, portions of its associated piping, and the main condenser are required to remain functional if credit is taken for deposition of iodine and if the SSE occurs. Consequently, the staff's past practice has been to classify these components as safety related and seismic Category I. In addition, 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.

Under Item II.E, "Classification of Main Steamlines in Boiling Water Reactors" of SECY-93-087, the staff stated that the classification of the MSL 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 MSL from the outmost isolation valve up to, but not including, the turbine stop valve are equivalent to the staff guidelines in Appendix A to SRP Section 3.2.2 and RG 1.29.

In SECY-93-087, the staff proposed for Commission approval, among other things, that (1) neither the main steam drain and bypass lines from the first valve up to the condenser inlet nor the piping between the turbine stop valve and the turbine inlet be classified as safety related or as seismic Category I; rather, these lines should be analyzed using a dynamic seismic analysis to demonstrate structural integrity under SSE loading conditions, and (2) the turbine stop, control, and bypass valves and the MSLs from the turbine control valves to the turbine meet all of the quality group and quality assurance guidelines specified in SRP Section 3.2.2, Appendix A.

For the purpose of providing a credit for iodine holdup and plateout, the staff's model requires that the main steam piping (including its associated piping to the condenser) and the condenser remain structurally intact following an SSE, so they can act as a holdup volume for fission products. By the term "structurally intact," the staff assumes the steamline will retain sufficient structural integrity to transport the relatively low flow ($\leq$ 10 $ft^3$/min) of MSIV bypass leakage throughout the steamlines and condenser. The staff considers, in its radiological consequence assessment, that the condenser is open to the atmosphere via leakage through the low-pressure turbine seals. Thus, it is only necessary to ensure that gross structural failure of the condenser will not occur. The amounts of iodine removal (removal coefficients) are plant design specific and, therefore, will be determined on a case-by-case basis.

The staff concludes that the ALWR vendor or COL applicant should provide reasonable assurance that the main steam piping from the outmost isolation valve up to the turbine stop valve, the MSIV leakage pathway (i.e., the drainline or bypass line) up to the condenser, and the main condenser will remain structurally intact and leaktight, so that they can act as a holdup volume for fission products during and following an SSE.

The staff's acceptance of the proposed elimination of the LCS and allowance of a higher MSIV leakage are subject to Commission approval of this issue as the staff recommended in the draft Commission paper dated February 10, 1994, "Source Term-Related Technical and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Design." The staff considers the two open issues identified in the DSER for MSIV leakage, and the vendor-specific item on plateout considerations to be closed.

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-Waster-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident"), and NUREG-0737 ("Clarification of TMI Action Plan Requirements"). Section 2.3.2 of Appendix B to Chapter 1 of the passive Requirements Document gives EPRI's proposed deviations from the design requirements for the PASS in several areas.

In Revision 3, EPRI listed the following deviations for the PASS in Section 2.3.2 of Appendix B to Chapter 1:

(1) relaxation of time requirements for sampling boron concentration and activity measurements

(2) elimination of hydrogen analysis of samples of containment atmosphere

(3) elimination of dissolved gas (hydrogen, oxygen, and total gases) and chloride analyses of reactor coolant samples

As discussed in greater detail in Section II.I of SECY-93-087, which is provided in Annex A of this appendix, the first deviation is acceptable because increasing the time requirement for sampling boron concentration to 8 hours and increasing the time requirement for activity measurement to 24 hours are justified by the passive ALWR design because, in the early phase of the accident, the information on reactivity and the condition of the core can be obtained from existing instrumentation. The Commission approved this position in a staff requirements memorandum (SRM) dated July 21, 1993, which is provided in Annex A of this appendix.

As discussed in greater detail in SECY-93-087, the second deviation is acceptable because safety-grade instrumentation inside the containment provides adequate monitoring for postaccident hydrogen. The staff has accepted this deviation at currently licensed plants. In addition, the Commission approved this position in its SRM dated July 21, 1993.

As discussed in greater detail in SECY-93-087, in the third deviation EPRI proposed to eliminate dissolved gas and chloride analyses of reactor coolant samples. This position is acceptable for BWRs because reactor vessel depressurization following the accident would render dissolved gas samples meaningless and, in other cases, the process sampling system could be used for taking dissolved gas samples. Also, chloride sampling in BWRs constitutes a secondary consideration, since these samples could be taken at a low pressure. This position was approved by the Commission in its SRM dated July 21, 1993.

For PWRs, however, elimination of dissolved gas sampling is not acceptable because certain accident sequences can occur in even a vented reactor vessel while the reactor coolant system is intact at reduced pressure and heat is removed by subcooled decay heat removal (as in the TMI-2 accident). For these cases, it would not be possible to evaluate concentrations of dissolved gases in reactor coolant to determine postaccident conditions existing in the reactor vessel. In the SRM dated July 21, 1993, the Commission approved the position that, while chloride will no longer be required, PWRs will still be required to determine the gross amount of dissolved gases (not necessarily a pressurized sample) as an acceptable means of satisfying the intent of 10 CFR 50.34 (f)(2)(viii) and Item II.B.3 of NUREG-0737. Further, the Commission agreed that the time for taking these samples can be extended to 24 hours after the accident.

### 2.4.1 Reactor Pressure Vessel Level Instrumentation for Pressurized-Water Reactors (PWRs)

Early versions of Appendix B to Chapter 1 of the passive Requirements Document included a Section 2.4.1, which proposed elimination of the reactor pressure vessel level instrumentation system (RVLIS) for a PWR on the basis of the design features of an ALWR to which the passive Requirements Document applies. In Section 3.5 of Chapter 3 of this report, the staff provides the background and its evaluation of this issue. In its letter dated December 6, 1991, EPRI stated that it would modify the Requirements Document for the evolutionary

plant to require a RVLIS. Revision 3 of the passive Requirements Document included a RVLIS for the passive plants, and the RVLIS is no longer listed as an optimization subject in Appendix B to Chapter 1 of the passive Requirements Document. This issue, which the staff stated in the DSER that it would evaluate during its review of an application for FDA/DC, is closed because a RVLIS is now required for passive plants.

2.5.1 Containment Leak Rate Testing

Section 2.5.1 of Appendix B of Chapter 1 of the passive Requirements Document proposes that the maximum interval between Type C leakage rate tests be 30 months instead of the 24 months currently required in Appendix J to 10 CFR Part 50. 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.

The staff finds that the EPRI's proposed maximum interval between Type C leakage rate tests deviates from the requirements of Appendix J to 10 CFR Part 50. However, the staff agrees that the proposed test interval will 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. In addition, the staff has extended the time interval for performing Type C leakage testing at currently operating plants beyond the Appendix J requirements on a case-by-case basis. In parallel with this review, the staff has developed proposed changes to Appendix J to 10 CFR Part 50 for all reactors. These changes were transmitted to the Commission in SECY-91-348. The proposed modifications of the regulation would have allowed the increased test interval and addressed other issues raised by EPRI. This proposed rulemaking has been withdrawn and is being reexamined under the Commission-approved program "Elimination of Requirements Marginal to Safety." In its letter of May 13, 1993, the Advisory Committee on Reactor Safeguards identified no significant safety penalty caused by this change to the maximum interval between Type C leakage rate tests and agreed with the proposed staff position. Therefore, the staff has recommended in SECY-93-087 that the Commission approve the position that until the rule change proceedings for Appendix J to 10 CFR Part 50 are complete, the maximum interval between Type C leakage rate tests for both evolutionary and passive plants should be 30 months rather than the 24-month maximum interval currently required in Appendix J. This was identified as an open issue in the DSER. The staff considers the open issue to be resolved and closed.

In addition, Revision 3, EPRI requested additional changes to the containment leak rate testing requirements of Appendix J to 10 CFR Part 50 as contained in

ANSI/ANS 56.8, 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 Annex A of this appendix and Sections 2.1.4 and 6.3 of Chapter 5.

2.5.2 Physically Based Source Term

Section 2.4.1 of Chapter 5 of the passive 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 technical information document TID-14844 ("Calculation of Distance Factors for Power and Test Reactor Sites") and 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 of the passive Requirements Document lists source term treatment for the passive ALWR as a plant optimization subject, and Section 1.2.3 of Chapter 5 defines the physically based source term for the passive ALWR plant.

In addition to the discussion in these sections, EPRI submitted to the NRC two technical reports titled "Passive ALWR Source Term," DOE/ID-10321, prepared by the U.S. Department of Energy (February 1991), and "Passive ALWR Containment Natural Aerosol Removal" prepared by PoleStar Applied Technology, Inc. (April 1993). 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 and passive 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 June 1992, the staff issued a draft staff report containing proposed revised accident source terms for light-water nuclear power plants to replace those of TID-14844, a technical information document published in 1962. The draft report was issued for public comment as NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plant." Since issuing NUREG-1465, the staff has received more than 200 comments from 20 U.S. and foreign organizations. The more significant public comments concern (1) the release fractions of the less volatile and nonvolatile elements such as barium, strontium, and cerium during the early-in-vessel and ex-vessel release phases, (2) chemical forms of iodine, and (3) fission-product release timing. The staff's release fractions of the less volatile and nonvolatile elements are about 10 to 100 times higher than those proposed by EPRI. Draft NUREG-1465 does not address the expected amounts of iodine in organic form in the containment following a design-basis accident (DBA), while EPRI specifies 0.15 percent of the core inventory iodine as the organic form.

The staff found in its radiological consequence assessments that the less volatile and nonvolatile elements play a less significant role, contributing less than a few percent of the overall radiological consequences resulting

from a DBA, than do the other major nuclides such as noble gases, iodine, and cesium. Therefore, the staff believes that the potential revisions to the release fractions of the less volatile and nonvolatile elements in draft NUREG-1465, if any, will not materially change its conclusions given in this FSER.

For the chemical forms of iodine, the staff found that the organic form of iodine may contribute the most significant offsite radiological consequences for the entire assumed duration of an accident (30 days). The staff used 0.25 percent of the core iodine inventory as the organic form in its review and assumed that it be immediately available for leakage from the containment to the environment following a DBA (see Section 2.1.9 of this report). Most of the remaining comments are all favorable regarding draft NUREG-1465. NRC contractors are reviewing the comments received and are expected to complete the review within the next several months. The staff is planning to issue the final version of NUREG-1465, complete with the fission-product removal mechanisms in 1994.

Draft NUREG-1465 addressed the current understanding of LWR severe-accident and fission-product behavior, including the quantity and chemical forms of fission products released into the reactor containment following a severe reactor accident as well as fission-product release timing. The report also contained a section on mechanisms for removing fission products from the reactor containment atmosphere following a severe reactor accident that was based on the realistic best estimates used in the NUREG-1150 ("Severe Accident Risk: An Assessment for 5 U.S. Nuclear Power Plants") analyses. To obtain more comprehensive information on removal mechanisms to be used in the licensing review, the staff is continuing to explore this area with the assistance of NRC contractors.

In reviewing the passive Requirements Document, the staff used the current insights from source term research as described in NUREG-1465 without waiting for more comprehensive information on fission-product removal mechanisms, which may become available from an NRC contractor's work, so that it could complete its review of the passive Requirements Document, supporting the ALWR integrated review milestones delineated in SECY-93-097, "Integrated Review Schedules for the Evolutionary and Advanced Light-Water-Reactor Projects." The staff's final positions used in the review of the source term-related issues are recommended for Commission approval in SECY-93-YYY. The staff believes that the final revision of NUREG-1465 will not materially change its conclusions stated in this FSER.

EPRI and the staff have resolved the major differences for all source term-related issues in principle except for a few minor differences (iodine chemical species for BWR, amount of gap activity, magnitude of low- and non-volatile fission products released and fission-product release timing). The staff believes that these minor differences can be resolved by the issuance of final NUREG-1465. The staff considers this open issue closed.

### 2.5.3 Hydrogen Control

The passive Requirements Document originally stated that the ALWR design would be capable of accommodating hydrogen generation resulting from the oxidation of 75 percent of the active fuel cladding, and for ALWRs with a non-inerted

atmosphere, the uniform hydrogen concentration would not be permitted to exceed 13 percent on a dry volume basis. EPRI stated that the potential effects of hydrogen burning must be evaluated and shown to be acceptable. For an inerted design, an inert atmosphere must be maintained.

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

- accommodate hydrogen equivalent to 100-percent metal-water reaction of the fuel cladding
- limit containment hydrogen concentration to no greater than 10 percent
- provide containment-wide hydrogen control (e.g., igniters, inerting) for severe accidents

In its June 26, 1990, staff requirements memorandum (SRM), the Commission approved the staff's position.

Under Item I.G of Commission paper SECY-93-087, the staff provides its position on hydrogen control for the passive ALWR designs; that is, that the passive ALWR designs be required to meet the same hydrogen control requirements (listed above) as the evolutionary designs. In Revisions 4 and 5 to the passive Requirements Document, Sections 2.4 and 6.5 of Chapter 5, were modified as follows:

- Under Top-Level Requirements for Accident Mitigation, Section 2.4.2.7: "The design shall ensure that containment integrity is maintained during an accident that releases hydrogen generated from 100 percent metal-water reaction of the active fuel cladding accompanied by hydrogen burning or the added pressure from postaccident inerting."
- Under Passive Autocatalytic Recombiner System Performance Requirements, Section 6.5.3.1: For passive PWRs a non-safety-grade passive autocatalytic recombiner (PAR) system will be provided that will ensure margin in the design for natural circulation mixing of the containment atmosphere by assuring that local hydrogen concentrations, except at the location of release, will not exceed 10 percent. EPRI also states, "The PAR system shall utilize components with demonstrated capability to combine hydrogen and oxygen under severe accident conditions at a sufficient rate and in a manner compatible with the passive ALWR concept."
- Under Inerting System Performance Requirements, Section 6.5.4.1.2: Passive BWRs will rely on an inerting systems to meet the combustible gas control requirements stated above. "The inerting system shall be capable of reducing the wetwell and drywell bulk oxygen concentrations in the containment to 3.5 volume percent in less than 4 hours and maintaining that concentration or less during normal plant operation." This requirement is designed to provide "margin to the ignitable hydrogen-oxygen ratios for the maximum range of design basis hydrogen generation." In addition, under Passive Autocatalytic Recombiner System Performance

Requirements, Section 6.5.3.1.2: "A non-safety grade PAR system shall be provided for a BWR inerted containment to prevent deinerting of either the drywell or the wetwell from net oxygen production during severe accident without reliance on any other combustible gas control devices."

The above new requirements satisfy the requirements of 10 CFR 50.34(f)(2)(ix), which the staff has determined are applicable to the passive ALWRs. Both plant designs are required to accommodate hydrogen generation equivalent to a 100-percent metal-water reaction of the fuel cladding. The passive PWR design will include a hydrogen control system that will limit hydrogen concentration to less than 10 percent and provide containment-wide hydrogen control for severe accidents. The passive BWR design will meet the requirement by ensuring that the postaccident atmosphere in the containment will not support hydrogen combustion. On this basis, the staff finds the proposed hydrogen control requirements to be acceptable. The staff notes, however, that while PARs used for severe-accident mitigation need not be safety grade, they would have to be safety grade if they are to be used for DBA mitigation. For BWRs, the PARs installed to control oxygen concentration for DBA mitigation must be safety grade. In addition, the staff is still reviewing the report submitted by EPRI by letter dated April 8, 1993, titled "Qualification of Passive Autocatalytic Recombiners (PARs) for Combustible Gas Control in ALWR Containments." The staff will evaluate the acceptability of the PAR system for combustible gas control during its review of an individual application for FDA/DC. The staff notes, however, that PARs are not the only way to acceptably meet the hydrogen control requirements. It still considers igniters to be acceptable. Therefore, the staff has not deleted from this report previous references to igniter systems and their acceptability.

2.5.4 Dedicated Containment Vent Penetration

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 June 26, 1990, SRM, 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 against 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.

Section 2.5.4 of Appendix B to Chapter 1 of the passive Requirements Document states that EPRI is opposed to providing a dedicated penetration for the purpose of future installation of a containment vent. EPRI's position is that the basis for the installation of a hardened vent in the Mark I containments has already been addressed in the ALWRs by design features that limit the potential for or permit the containment to accommodate the challenge of overpressurization. EPRI suggests that in lieu of a dedicated containment vent, containment venting using existing plant equipment be included in plant emergency procedures along with other accident management strategies.

Because of the stage of design development, the staff has insufficient information to determine if a containment vent is necessary for passive plant designs. Therefore, it will address this issue during its review of an individual application for FDA/DC. This item is discussed further under Item I.K of Annex A of this appendix.

2.5.5 Anticipated Transients Without Scram

In Section 2.5.5 of Appendix B to Chapter 1 of an earlier version of the passive Requirements Document, EPRI stated that the document fully addressed the regulatory requirements of 10 CFR 50.62 and GDC 26 regarding reduction of the probability of an anticipated-transient-without-scram (ATWS) event and enhancement of the mitigation capability if such an event occurred in the passive 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, the automatic initiation of the standby liquid control system (SLCS) as required by 10 CFR 50.62 was not necessary for the passive BWR.

However, in its December 6, 1991, letter, EPRI stated it had determined that automatic actuation of the SLCS was appropriate for the evolutionary BWR design, and that it was modifying the Requirements Document for evolutionary plants to reflect that position. EPRI no longer considered this to be a plant optimization subject for the evolutionary plant design.

In the DSER for Appendix B to Chapter 1 of the passive Requirements Document, the staff stated that although EPRI had not submitted its position regarding the automatic actuation of the SLCS for passive designs, the staff expected EPRI to provide design requirements for the passive designs that were consistent with those for the evolutionary designs. It stated that it would evaluate EPRI's position regarding this issue when it is submitted. This was identified as an open issue. However, in Revision 4 of the passive Requirements Document, EPRI revised Sections 4.5.2.4 and 4.5.3.5.1 of Chapter 5 and Section 8.7.2 of Chapter 10 to require automatic actuation of the SLCS on a signal(s) that is unique to an ATWS event. Section 2.5.5 of Appendix B to Chapter 1 has been deleted as a plant optimization subject. Therefore, the open issue is closed. The staff also discusses this issue in this report under Item I.B of Annex A of this appendix and Section 4.5 of Chapter 5.

2.5.6 Safe Shutdown

Passive ALWR designs use passive heat removal systems for decay heat removal (DHR). They are restricted by the inherent capability of the passive heat removal processes and cannot reduce the temperature in the reactor coolant system (RCS) 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 simplified BWR. Even though active shutdown cooling systems are available to bring the reactor down to cold shutdown or refueling conditions, these active residual heat removal systems are not safety grade and do not comply with the guidance of RG 1.139, "Guidance for Residual Heat Removal," or Branch Technical Position RSB 5-1 (NUREG-0800).

Section 2.5.6 of Appendix B to Chapter 1 of the passive Requirements Document states that it is not necessary for the passive safety systems to be capable

of achieving cold shutdown because EPRI believes that the passive DHR systems have an inherently high long-term reliability. EPRI states that the passive ALWR designs will employ a redundant safety system, for both the hot standby and long-term cooling modes. EPRI originally defined safe shutdown as 204.4 °C (400 °F); in Revision 4 of the passive Requirements Document, it redefines safe shutdown as 215.6°C (420°F). EPRI indicates that it meets GDC 34 because redundant passive DHR systems can operate at full RCS pressure and place the reactor in the long-term cooling mode immediately after shutdown. In the DSER, the staff stated that it was evaluating the issue pertaining to the definition of safe shutdown and identified this as an open item.

As discussed in Section 3.3 of Chapter 5 of this report, the staff has completed its evaluation of the issue pertaining to the definition of safe shutdown. As stated there, the staff concludes that cold shutdown is not the only safe, stable shutdown condition to maintain the fuel and reactor coolant boundary within acceptable limits, and that 215.6 °C (420 °F) proposed by EPRI as a safe, stable shutdown condition is acceptable on the basis of acceptable passive safety system performance and acceptable resolution of the regulatory treatment of non-safety systems (the "RTNSS" issue). As discussed in Section 2.2 of Chapter 3 of this report, EPRI has proposed an acceptable process for resolving the RTNSS issue. Any non-safety systems that are required for maintaining long-term stable shutdown conditions will be included as part of the RTNSS evaluation. If a non-safety system is found risk significant, it will be subject to proper regulatory oversight commensurate with its reliability/availability mission and risk significance.

The staff's technical position regarding EPRI's proposal that 215.6 °C (420 °F) be defined as a safe shutdown condition is given in greater detail in Annex B of this appendix. The Commission approved this position in its staff requirements to memorandum (SRM) dated June 30, 1994. The open issue is closed.

# 3 GENERIC SAFETY ISSUES

## 3.1 Introduction

10 CFR 52.47(a)(1)(iv) requires that applications for design certification contain proposed technical resolutions of those unresolved safety issues and medium- and high-priority generic safety issues which are identified in the version of NUREG-0933 current on the date six months prior to application and which are technically relevant to the design.

In Section 3.1 of Appendix B to Chapter 1 of the passive Requirements Document, EPRI addresses 21 generic safety issues for which there was no generic resolution as of January 1990 in the NRC generic issues management control system (GIMCS) and that EPRI determined were technically relevant to the ALWR passive plant designs using the criteria in NUREG-1197, "Advanced Light Water Reactor Program - Program Management and Staff Review Methodology." These issues* are as follows:

| | | | | | | |
|---|---|---|---|---|---|---|
| I.D.3 | 15 | 70 | 83 | 94 | 121 | HF 4.4 |
| B-17 | 29 | 75 | 84 | 105 | 128 | HF 5.1 |
| C-8 | 57 | 79 | 87 | 113 | 130 | HF 5.2 |

Since January 1990, 16 of the above issues (I.D.3, C.8, 29, 70, 75, 79, 84, 87, 94, 113, 121, 128, HF 4.4, HF 5.1, and HF 5.2) have been technically resolved by the staff. The staff will evaluate the proposed elements for resolving the remaining 5 issues (B-17, 15, 57, 83, and 105) on the basis of the final agency issue resolution.

A note to Table B.3-1 in Appendix B to Chapter 1 of the passive 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-56, B-61, B-64, 23, 106, and 135 were not technically relevant to the ALWR design. EPRI, however, did not give the reasons why it determined specific issues were not technically relevant to the ALWR passive plant designs. In addition, although the passive Requirements Document does not propose elements of resolution for Issue II.J.4.1, it includes commitments to comply with the revised regulations that form the basis for the staff's identified resolution of the issue. In the DSER for Appendix B to Chapter 1, the staff concluded that EPRI should explicitly give its reasons why it determined specific issues were not technically relevant in a future revision of the passive Requirements Document. A discussion of these issues is included in the staff's evaluation of generic issues in Section 3.2, below.

The vendors will be required to address, in accordance with 10 CFR 52.47(a)(1)(iv), any that are applicable to the ALWR passive plant designs and that are prioritized as high or medium.

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*Table 3B.1 in this appendix gives the titles of the issues, their NUREG-0933 status, the section of Appendix B to Chapter 1 of the passive Requirements Document where each one is addressed, and other remarks.

Irrespective of the specific issues addressed in this evaluation and in the passive Requirements Document, 10 CFR 52.47(a)(1)(iv) requires that applications for FDA/DC 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 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 passive 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 Issue I.D.3. This issue was resolved in September 1993 with no additional requirements for existing or future plants.

### 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, "List of Generic Safety Issues," of Appendix B to Chapter 1 of the passive Requirements Document, EPRI states that Issue I.D.5(3) was determined, through the NUREG-1197 screening process, not to be 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 passive 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. While 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 dictate that this issue is otherwise applicable to ALWR passive plant designs, the staff will expect the FDA/DC applicant to address its resolution for staff evaluation.

3.2.3 II.H.2, Obtained Technical Data on the Conditions Inside the TMI-2 Containment Structure

Issue: TMI Action Plan Item II.H.2 was devoted to the efforts necessary to acquire and record information during the cleanup of the TMI-2 facility.

EPRI Proposal: In the note to Table B.3-1 of Appendix B to Chapter 1 of the passive Requirements Document, EPRI states that Issue II.H.2 was determined through the NUREG-1197 screening process not to be technically relevant to the ALWR design.

Staff Evaluation: The staff concludes that data acquisition and recording from the TMI-2 cleanup was an issue applicable to a single facility and not to future plants. Therefore, EPRI's position, that Issue II.H.2 is not technically relevant to the ALWR design, is acceptable.

3.2.4 II.J.4.1, Revised 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 of Appendix B to Chapter 1 of the passive Requirements Document, EPRI states that Issue II.J.4.1 was determined, through the NUREG-1197 screening process, not to be technically relevant to the ALWR design rather, it addresses operational considerations. Because 10 CFR Part 21 and 50.55(e) contain both design and other considerations, EPRI's commitment in Table B.1-1 in Appendix B to Chapter 1 (Rev. 1) of the passive Requirements Document shows an ALWR Program commitment to comply with the portions of 10 CFR Part 21 and 50.55(e) that are design related.

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) were made in Table B.1-1 of Revision 1 of the passive Requirements Document, which was published in April 1991 before the rule changes were issued. In the DSER for Appendix B to Chapter 1, the staff stated that the commitments would satisfy its primary concerns regarding this issue provided EPRI confirmed that the commitments included the most recent rule changes. This was identified as a confirmatory issue. EPRI has stated that the ALWR Program assumes that designers will comply with current regulations. Dates or Federal Register citations are not included in Table B.1-1 of the passive Requirements Document, but the intent is clear that current regulations are to be used. In the DSER, the staff also stated that EPRI should make appropriate revisions to the passive Requirements Document to address the apparent error noted in the "EPRI Proposal" discussion above. This was identified as an open issue. The staff considers both the confirmatory and open issues resolved in that during its

review of an individual application for FDA/DC, the staff will ensure that the concerns of Issue II.J.4.1 have been satisfactorily addressed by the applicant.

The discussion of Issue II.J.4.1 in NUREG-0933 has not been updated to reflect the recent rule changes. Should the final agency resolution of this issue dictate other actions for ALWR passive plant designs, the staff will expect the FDA/DC applicant to address those actions for staff evaluation.

3.2.5 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, 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 passive Requirements Document states:

> The ALWR plant design requires the automation of the initiation of protection and safety systems and, in addition, the automation of the operation of these systems precludes the need for operator action for at least 72 hours after the initiation of an event, plus licensing design basis (LDB) assumptions, to meet regulatory requirements. 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 states:

- Section 2.3.2.9 of Chapter 1, specifies that the passive ALWR shall permit increased operator response time over existing LWRs. For transients and accidents analyzed under the initiating event plus single failure LDB assumptions, no credit for operator action shall be necessary to meet regulatory limits until at least 72 hours following the initiating events.

- Sections 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, for investment protection.

- 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 passive ALWR PWR. The passive PWR safety injection system (PSIS) maintains inventory by gravity feed of water from an in-containment refueling water storage tank (IRWST) to the reactor vessel. (See Chapter 5, Sections 3.2 and 5.1. Specifically, Section 3.2.3 of Chapter 5 requires that the PSIS shall be designed such that no system realignment is required to maintain system operation to completion.)

Section 3.10.2.4.3 states:

> The passive 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 passive plant.

Staff Evaluation: On the basis of EPRI's response to the staff's DSER dated September 1991 for the evolutionary Requirements Document, the staff concludes that EPRI has provided the technical basis and specific requirements that support its assertion that the quantitative standards and the process requirement in the passive Requirements Document are considered to be significantly

more limiting than those in ANSI/ANS 58.8-1984, "Time Response Design Criteria for Safety-Related Operator Action." EPRI has further committed to satisfy NRC requirements concerning automation of safety-related operator actions and operator response times. The NRC has not yet endorsed ANSI/ANS 58.8; however, the standard is being revised and it is expected that the revised standard will be endorsed. Since the planned revision to ANSI/ANS 58.8 will not 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 passive plant designs.

3.2.6 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 passive Requirements Document, EPRI states that Issue B-55 was determined, through the NUREG-1197 screening process, not to be 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. In the DSER for Appendix B to Chapter 1, the staff requested that EPRI document its reasons for determining that this issue was not applicable to the ALWR passive plant. This was identified as an open issue. The staff also requested that EPRI confirm the staff's assumption that Target Rock safety valves would not be used in ALWR passive plants. This was identified as a confirmatory issue. In a letter dated September 23, 1992, EPRI responded to these issues by pointing out that Section 12.2.2.3 in Chapter 1, Revision 1 (which is 12.2.1.11 in Revision 4) reads as follows: "Pilot actuated, power operated relief valves will not be used in ALWR applications for overpressure protection or for system depressurization service." Because EPRI has stated that these valves will not be used in passive ALWR designs, the staff concludes that Issue B-55 is not applicable to the passive Requirements Document. The open and confirmatory issues are closed.

3.2.7 B-56, Diesel Reliability

Issue: A review of licensee event reports showed that onsite emergency diesel generators (EDGs) at operating nuclear power plants were demonstrating an average starting reliability of 0.94 per demand. (NRC's goal for new plants as expressed in RG 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," is 0.99 per demand.) A study has been completed and the more significant causes of diesel generator unreliability and recommended corrective action are documented in NUREG/CR-0660, "Enhancement of On-Site Emergency Diesel Generator Reliability." Improving the starting reliability of onsite EDGs will reduce the probability of events that could escalate into a core-melt accident, and thus, could effect an overall reduction in public risk.

EPRI Proposal: In the note to Table B.3-1 of Appendix B to Chapter 1 of the passive Requirements Document, EPRI states that Issue B-56 was determined, through the NUREG-1197 screening process, not to be technically relevant to the ALWR design. However, Table B.1-2 in Appendix B to Chapter 1 of the passive Requirements Document shows an ALWR Program commitment to comply with the intent of RG 1.108, Revision 1 (discussed further in Section 4.23 of Appendix B to Chapter 1 of the passive Requirements Document).

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-56 is listed as a Code (or Category) 4 item, that is, an issue that is applicable only to operating plants. Issue B-56 does not appear to have any effect on ALWR plant design, schedule, or construction costs and is, therefore, considered by the staff to be not applicable using the definitions of Appendix A to NUREG-1197. In the DSER for Appendix B to Chapter 1, the staff recommended that EPRI document a rationale explaining why this issue is not applicable to the passive ALWRs and identified this as an open issue. Because any additional requirements for diesel generators (including periodic testing) would be determined in accordance with the RTNSS process, the staff will evaluate diesel reliability during its review of an individual application for FDA/DC. Therefore, the open issue is closed.

3.2.8 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 TS. 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 of Appendix B to Chapter 1 of the passive Requirements Document, EPRI states that Issue B-61 was determined, through the NUREG-1197 screening process, not to be technically relevant to the ALWR design.

Staff Evaluation: Appendix A to NUREG-1197 identifies issues 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. Issue B-61 does not appear to address plant design and is, therefore, considered by the staff to be not applicable using the definitions of Appendix A to NUREG-1197. This was identified as an open issue in the DSER for Appendix B to Chapter 1. However, since B-61 is not a design issue and not relevant to the ALWR design, this issue is closed.

3.2.9 B-64, Decommissioning of Reactors

Issue: Decommissioning is defined as the orderly retirement of a nuclear facility from service and the safe disposition of the remaining radioactivity.

EPRI Proposal: In the note to Table B.3-1 of Appendix B to Chapter 1 of the passive Requirements Document, EPRI states that Issue B-64 was determined through the NUREG-1197 screening process not to be technically relevant to the ALWR design. However, EPRI has included requirements in several sections throughout the Requirements Document that are intended to facilitate the decommissioning process.

Staff Evaluation: This issue was closed in 1988 with the development of decommissioning rules and their subsequent inclusion in 10 CFR. NUREG-0933 identified this issue as not applicable to future plants. Therefore, the staff concludes that EPRI's position, that Issue B-64 is not technically relevant to the ALWR design, is acceptable.

3.2.10 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 it 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 3.3.1 of Appendix B to Chapter 1 of the passive Requirements Document states that Issue C-8 is addressed in Section 2.3.1 of Appendix B as an ALWR optimization subject.

Staff Evaluation: As discussed in Section 3.1 of this appendix, since Issue C-8 is not included in Appendix B to NUREG-0933, the staff determined that it is not applicable to future plants. It should also be noted that the staff has resolved Issue C-8 in March 1990 with no additional requirements.

The optimization subject of MSIVLCS is discussed in Section 2.3.1 and under Item II.E of Annex A of this appendix.

3.2.11 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 (that is, 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 passive Requirements Document.

Staff Evaluation: In Section 3.3 of Chapter 4 of this report, the staff concludes that the EPRI-proposed requirements for reactor pressure vessel supports are acceptable at this time. However, Issue 15 has not yet been generically resolved by the staff and has a high safety priority. Should final agency resolution dictate additional actions for ALWR passive plant designs, the staff will expect the FDA/DC applicant to address the resolution of Issue 15 for staff evaluation in accordance with 10 CFR Part 52.

In addition, as discussed in Section 3.3 of Chapter 1 of this report, the staff will review the ALWR plant designs for a 60-year life. Should it request such a review, the FDA/DC applicant will be required to provide information and programs to address the effects of radiation and other phenomena on the reactor pressure vessel supports over a 60-year life.

3.2.12 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: In the note to Table B.3-1 of Appendix B to Chapter 1 of the passive Requirements Document, EPRI states that Issue 23 was determined, through the NUREG-1197 screening process, not to be technically relevant to the ALWR design.

Section 3.4.2.8 of Chapter 3 of the passive Requirements Document states that the RCPs will not use rotating shaft seals. This section indicates that canned motor pumps, or other pumps of a proven design without seals, will be used in the ALWR passive plant.

Staff Evaluation: Appendix A to NUREG-1197 identifies issues the staff determined were not applicable to the ALWR Program according to six categories. Issue 23 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.

On the basis of the information in Section 3.4.2.8 of Chapter 3 of the passive Requirements Document, the staff has determined that Issue 23 meets the definition in Appendix B to NUREG-1197 of a not applicable generic safety issue. Furthermore, EPRI's proposed approach appears to adequately address the main concern of Issue 23. The passive design will use canned pumps that do not use seals. Therefore, the issue is not applicable to this design.

3.2.13 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 LOCA 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 of resolution for this issue are included in Section 3.1.1 of Appendix B to Chapter 1 of the passive 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, "Threaded Fastener Experience in Nuclear Power Plants," and NUREG-1095, "Evaluation of Responses to IE Bulletin 82-02") and by industry (Institute of Nuclear Power Operations SOER 84-5 and EPRI NP-3784). Also mentioned is the cooperative industry effort between the Atomic Industrial Forum/Metal Properties Council Task Group on Bolting and EPRI to resolve the bolting issue (which resulted in EPRI NP-5769). The 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 intention to revise the SRP (NUREG-0800) to provide criteria for evaluating the bolting practices proposed by applicants for new plants is also mentioned.

Section 3.1.1.4.2 provides requirements for all ALWRs regarding Issue 29. These requirements are pulled together from various sections of Chapter 1 of the passive Requirements Document:

- 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 the use of 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 practical, 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 of the passive Requirements Document 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 support 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 shall 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 passive 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 for the 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 passive Requirements Document addresses all elements considered for resolving Issue 29 and that this issue is resolved for the ALWR.

Staff Evaluation: The staff technically resolved Issue 29 in October 1991 with the issuance of Generic Letter (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.

In the DSER for Appendix B to Chapter 1, the staff stated that even though the EPRI proposal discussed above appeared to have met most of the recommendations in GL 91-17, the staff was 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, "Good Bolting Practices, A Reference Manual for Nuclear Power Plant Maintenance Personnel," 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 pertaining to 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 passive Requirements Document.

The three concerns discussed above were identified as open issues in the DSER. In addition, the NRC staff concluded in the DSER that since the passive Requirements Document did not provide any commitment to GL 91-17 or sufficient design details (e.g., the selection of bolting materials and of lubricants and leak sealants), the staff would evaluate resolution of this issue during its review of individual applications for FDA/DC and this issue was also considered to be a utility- or vendor-specific item.

In a letter dated September 29, 1992, EPRI explained that the basis for the technical resolution of Issue 29 is found in Section 12.7 of Chapter 1. The NRC staff reviewed Section 12.7 and found it gives the same technical basis as that in GL 91-17. The requirements in Section 12.7 will ensure that bolting and fastener materials are selected from materials proven in service, and thus bolts and fasteners will perform satisfactorily in service. The staff concludes that the vendor-specific and these open issues identified in the DSER have been satisfactorily resolved and are closed.

### 3.2.14 57, Effects of Fire Protection System Actuation on Safety-Related Equipment

Issue: This issue addresses fire protection system (FPS) actuation that has 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 an information notice to alert licensees and provided recent examples in which FPS actuation 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 passive Requirements Document. Several proposed elements include but are not limited to the following:

- Preaction sprinkler systems will be used where the undesirable consequences of leakage or inadvertent operation are high.
- Seismic sensitive relays will not be used in carbon dioxide systems.
- In the selection and maintenance of fire detectors, specific consideration of the effects of humidity, air velocity, temperature, air pollution, radiation, and air pressure will be required. In addition, the detection circuit will be designed so that the failure, removal, or replacement of the detector will not affect the performance of the detection loop.

Staff Evaluation: In the DSER for Appendix B to Chapter 1, the staff identified this as an open issue because Issue 57 had not yet been generically resolved by the staff and it had a medium safety priority. Although the staff considers EPRI's proposed resolution to be acceptable in intent, it will evaluate the effect of inadvertent FPS actuation on safety-related equipment during its review of an individual application for FDA/DC. Therefore, the open issue is closed.

3.2.15 70, Powered-Operated Relief Valve and Block Valve Reliability

Issue: Power-operated relief valves (PORVs) and block valves were originally designed as non-safety 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 passive Requirements Document.

Staff Evaluation: In the DSER for Appendix B to Chapter 1, the staff stated that it had not yet completed its review of EPRI's proposed resolution and identified this an open issue. The staff generically resolved Issue 70 in June 1990 and issued GL 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,'" (requiring TS revisions at PWRs with PORVs and block valves). It has also drafted revisions of the SRP (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. The staff has found that Section 3.3.2 of Chapter 3 and Section 5.4 of Chapter 5, as summarized in Section 3.5.4, "Depressurization System for PWRs," in Appendix B to Chapter 1 of the passive Requirements Document implicitly contain the commitment to LTOP. The staff evaluation is documented in Section 3.3 of Chapter 3 of this report.

It should be noted that the passive ALWR design relies on the depressurization system (DPS) for safety-grade depressurization of the reactor coolant system (RCS) and venting of noncondensible gases from the pressurizer. Sections 4.4.2 and 5.4.2 of Chapter 5, respectively, specify the BWR and PWR DPS performance requirements for depressurization capabilities. Section 6.6.4.3 of Chapter 5 specifies that the DPS will be capable of reducing RCS pressure to 250 psig or less before reactor vessel meltthrough during a severe accident. The staff finds this acceptable because depressurization to below 250 psig will reduce the potential for high-pressure core melt ejection. The open issue is closed.

3.2.16 75, Generic Implications of ATWS 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 (1 part), resolution of an unresolved safety issue (1 part), and staff plans to revise a regulatory guide (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 passive Requirements Document.

Staff Evaluation: The staff has resolved Issue 75 and has issued GL 83-28 and IE Bulletins 83-01 and 83-04. In the DSER, however, it identified this as an open issue because it had not yet completed its review of EPRI's proposed elements of resolution for Issue 75.

This issue primarily addresses the action items related to the reactor protection system proposed by the NRC task force as a result of studies on failure to scram automatically in the Salem plant. It deals with the staff action items on quality assurance and general operating criteria of the reactor trip system and the licensee action items described in GL 83-28 on the issues of post-trip review, equipment classification and vendor interface, post-maintenance testing, and reactor trip system reliability. The passive Requirements Document does not require compliance with GL 83-28 in Table B.1-2 or elsewhere. The staff will evaluate the specific design during its review of an individual application for FDA/DC to ensure that all aspects of the ATWS issue have been properly considered, including compliance with GL 83-28. Therefore, the open issue is closed. See Item I.B of Annex A of this appendix for additional discussion of ATWS issues.

### 3.2.17 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 passive 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: The staff's efforts to resolve Issue 79 were based on a review of a Babcock & Wilcox (B&W) NCC analysis and the results of an NCC analysis performed by an NRC contractor, both of which were performed for the B&W 177 fuel assembly RV (B&W 177).

The staff noted limitations associated with the complicated analysis techniques and some calculated stresses that were as high as 98 percent of the ASME Code allowable. 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 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/hr 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.

In the DSER for Appendix B to Chapter 1, the staff concluded that for the above reasons and since the passive Requirements Document did not provide sufficient commitments regarding the cooldown rate to be used in ALWR NCC analyses, EPRI's proposed resolution of this issue would be acceptable if the passive Requirements Document included a requirement to analyze an NCC event from 100-percent reactor power to cold shutdown as part of the RV ASME Code design requirements and the analysis (1) assumed 30 NCC events for the 60-year design life of the RV and (2) used the maximum allowable cooldown rate specified by ALWR TS. This was identified as an open issue. In Section 6.2.1 of Chapter 4 of the passive Requirements Document, EPRI has committed to design the plant for at least 30 NCC cycles during the 60-year plant life consistent with the RCS heat removal capability. The staff finds this acceptable and the open issue is closed.

3.2.18 83, Control Room Habitability

Issue: This issue addresses the significant discrepancies found 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 TS.

EPRI Proposal: Section 3.5.1.4 of Appendix B to Chapter 1 of the passive Requirements Document states that the resolution of Issue 83, in general, is not design related. EPRI considers the concerns to be operating plant activities in the areas of maintenance, testing, installation, and training and, therefore, beyond the scope of the passive Requirements Document.

Section 1.2.1.3 of Chapter 5 of the passive Requirements Document requires that the control room complex be designed to limit leakage from potentially radioactive external environments to the internal environment of the control room complex. Section 4.36.4 of Appendix B to Chapter 1 states that control room habitability will be ensured by a non-safety-grade heating, ventilating, and air conditioning (HVAC) system with charcoal filtration and powered from non-safety onsite and offsite power sources. A passive control room pressurization system will be provided to minimize inleakage and keep operator doses below the limits specified in GDC 19 and SRP Section 6.4, Revision 2.

Staff Evaluation: In the DSER for Appendix B to Chapter 1, the staff concluded that the passive Requirements Document should address the staff's conclusions regarding the control room habitability systems addressed in Section 6.5 of Chapter 5 of the DSER and should state that the FDA/DC applicant needs 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. This was identified as an open issue. The staff evaluation of EPRI's proposed control room pressurization system is documented in SECY-93-087, SECY-94-084, and Section 6 of Chapter 5 of this report. Furthermore, the staff concurs with EPRI's position that this issue is not design related. Instead, it has to do with implementation such as maintenance, testing, installation, and training. Finally, the staff's evaluation of the non-safety-grade HVAC system will be in accordance with the process for the regulatory treatment of non-safety systems in passive plants which is described in SECY-94-084 in Annex B of this appendix. The staff will evaluate this process during its review of an individual application for FDA/DC when vendor- or utility-specific information is available. Therefore, this open issue is closed.

3.2.19 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 & 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 the 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 passive Requirements Document.

Staff Evaluation: The staff has generically resolved Issue 84, establishing no new requirements for current operating plants or future plants.

3.2.20 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 in operating and evolutionary BWRs 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 of passive plant designs.

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 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 passive 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 passive plant designs. EPRI gives three 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 passive Requirements Document. Second, EPRI states that it will take full advantage of its program to develop an MOV

performance prediction methodology. Third, EPRI states that the safety-related systems that normally use MOVs for isolation are not present in the ALWR passive plant designs.

Staff Evaluation: The staff generically resolved Issue 87 in December 1991 with no additional requirements.

However, the staff does not agree with EPRI's position that the MOV issue should be considered resolved for the ALWR passive plant. For example, as discussed in Section 12.2.2 of Chapter 1 of this report, the specific requirements sections of the passive Requirements Document have not fully resolved NRC staff concerns regarding the performance of MOVs in the ALWR passive 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 passive 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 item during its review of individual applications for FDA/DC in accordance with the above position and the staff positions on MOVs stated in Section 12.2.2 of Chapter 1 of this report.

3.2.21 94, Additional Low-Temperature Overpressure Protection for Light-Water Reactors

Issue: Unresolved Safety Issue A-26, which addressed low-temperature overpressurization, was resolved in September 1978 with a revision to SRP Section 5.2. This resolution affected all operating and future PWRs and required PWR licensees to implement procedures to reduce the potential for overpressure events and to install equipment modifications to mitigate such events. Current guidance on this issue can be found in SRP Section 5.2.2, "Overpressure Protection," and Branch Technical Position RSB 5-2 (NUREG-0800). From 1979 to mid-1983, 12 pressure transients were reported, 2 of which (at the same plant) exceeded the TS limit (415 psig below 355 °F) by about 700 psig and 351 psig, respectively. The number of overpressure transient events suggested potential weaknesses in the existing overpressure protection criteria or their implementation that warranted further staff investigation.

EPRI Proposal: Proposed elements for resolving this issue are included in Section 3.3.2 of Appendix B to Chapter 1 of the passive Requirements Document.

Staff Evaluation: In the DSER, the staff stated that although Issue 94 had been generically resolved with the issuance of GL 90-06, it had not yet completed its review of EPRI's proposed elements of resolution. This was identified as an open issue. GL 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' required a revision to the TS for overpressure protection at the affected plants. In addition, the staff has established guidance for resolving the low-temperature overpressure protection (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.'" The staff will evaluate the adequacy of the specific design for LTOP during its review of individual applications for FDA/DC. Therefore, the open issue is closed.

3.2.22 105, Interfacing System Loss-of-Coolant Accident at LWRs (integrates Issue 96, Residual Heat Removal Suction Valve Testing)

Issue: The common concern in Issues 105 and 96 is that 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 3.5.2.4 of Appendix B to Chapter 1 of the passive Requirements Document states: "The ALWR passive plant is being required to satisfy all of the current regulatory requirements which are intended to assure an acceptable level of isolation capability." It states that all ALWR passive plants are required to satisfy additional items, including the following:

- 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 RCS URS requirement, the plant designer will determine by evaluation that 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 (LOCA).

- Each of the high- to low-pressure interfaces (viz., those between those systems normally exposed to the RCS or RCS pressure and the interfacing lower pressure system) will include the following protection 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

An additional requirement for passive BWRs is that the RWCU system design pressure be at least as high as that of the reactor pressure vessel.

An additional requirement for passive PWRs is that the passive decay heat removal (PDHR) system be designed for full RCS design pressure and temperature.

Staff Evaluation: In a staff requirements memorandum (SRM) dated June 26, 1990, the Commission approved the staff's position on intersystem LOCA stated in SECY-90-016 with the provision that all elements of the low-pressure system are considered (e.g., instrument lines, pump seals, heat exchanger tubes, and valve bonnets). On April 2, 1993, the staff issued SECY-93-087 (Annex A of this appendix), which recommends that the passive plants be reviewed for compliance with the intersystem criteria approved in the above SRM. As

discussed in SECY-90-016, the staff concluded that designing low-pressure systems to withstand full RCS pressure, to the extent practicable, is an acceptable means of resolving the intersystem LOCA issue. This conclusion is essentially embodied in Section 2.2.14.1 of Chapter 3 of the passive Requirements Document.

In SECY-90-016, the staff also recommended that for systems that have not been designed to withstand full RCS pressure, evolutionary ALWRs provide

- the capability for leak testing 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 passive Requirements Document.

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. Regardless of the URS of low-pressure system piping and components, for additional assurance that these systems will not be exposed to RCS pressure, the staff will evaluate inservice testing programs and TS during its review of individual applications for a COL to ensure that all pressure isolation valves are identified and adequately leak tested in accordance with applicable plant-specific TS. In the DSER, this issue was considered to be a part of Policy Issue I.F. While leak testing of pressure isolation valves is briefly mentioned in Policy Issue I.F, the staff considers this to be a vendor-specific item. During its reviews of individual applications for FDA/DC, the staff will (1) review the standard safety analysis reports to determine if all pressure isolation valves are identified and (2) review the applicable TS to determine if all pressure isolation valves will be leak tested in accordance with staff positions. See Policy Issue I.F in Annex A of this appendix for additional discussion of this topic.

Since the passive Requirements Document does not specifically identify all interfacing systems, the staff will evaluate other interfacing system designs during its reviews of individual applications for FDA/DC. The staff intends to ensure that all elements of low-pressure systems have been considered in accordance with approved requirements, particularly those systems to be evaluated under Section 2.2.14.2 of Chapter 3 of the passive Requirements Document.

Issue 105 has not yet been generically resolved by the staff. Should final agency resolution of this issue dictate additional actions for ALWR passive plant designs, the staff will expect the FDA/DC applicant to address resolution of Issue 105 for staff evaluation.

3.2.23 106, Piping and Use of Highly Combustible Gases in Vital Areas

Issue: Several kinds of combustible gases are used during normal operation of nuclear power plants, usually in limited quantities and for relatively short periods. The accumulation of combustible or explosive mixtures of gas represents a threat to the safety of the plant because of the potential for disabling safety-related equipment if the combustible gases are inadvertently ignited. This issue addresses the safe use of relatively small amounts of combustible gases on site in normal process systems.

EPRI Proposal: In the note to Table B.3-1 of Appendix B to Chapter 1 of the passive Requirements Document, EPRI states that Issue 106 was determined, through the NUREG-1197 screening process, not to be technically relevant to the ALWR design.

Table B.1-2 in Section 1 of Appendix B to Chapter 1 of the passive Requirements Document shows an ALWR Program commitment to comply with SRP Section 9.5.1, Revision 3, "Fire Protection Program."

Staff Evaluation: In the DSER for Appendix B to Chapter 1, the staff identified resolution of Issue 106 as an open issue because it had not yet been generically resolved and had a medium safety priority. Issue 106 was generically resolved in June 1990, with additional requirements given in GL 90-06, "Piping and the Use of Highly Combustible Gases in Vital Areas." The staff's current position is that an acceptable approach for addressing the aspect of this issue related to volume control tank and generator applications is for the designer to ensure compliance with the following:

(1) current SRP Section 9.5.1 with the modifications to BTP CMEB 9.5-1, Part C.5.d(5) as outlined in (3)(a) below

(2) EPRI NP-5283-SR-A with respect to those portions of the guidelines involving hydrogen that do not deal specifically with hydrogen water chemistry (HWC)

(3) the following general guidance:

    (a) Modified BTP CMEB 9.5-1, Part C.5.d(5) indicates: Hydrogen lines in safety-related areas should follow the guidance of Regulatory Guide 1.29, "Seismic Design Classification," Section C.2. The lines should (a) 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 volume percent or (b) be sleeved with the outer pipe vented directly to the outside.

    (b) The hydrogen piping and components should be located to reduce risk (e.g., by reducing piping length and proximity to safety-related equipment).

    (c) 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 used to limit hydrogen releases to a leak/break.

(d) Equipment and controls to mitigate the consequences of a hydrogen fire/explosion should be accessible and remain functional during the event.

(e) Design features and/or administrative controls should be provided to isolate the hydrogen supply if normal building ventilation is lost (e.g., from building isolation caused by operation actions following a hydrogen fire/explosion in auxiliary building).

(f) Back flow to a leak/break of hydrogen contained in components (e.g., generator) should be included in evaluating the consequences of leaks/breaks and measures taken to mitigate these consequences.

(g) Threaded joints in the hydrogen distribution lines within safety-related areas should be back welded.

(4) the recommendation that safety-related equipment not be located in the turbine building because of the hazards associated with hydrogen fires and explosions and large oil fires and the large uncertainties in estimating the consequences

The staff will evaluate, during its review of an individual application for FDA/DC, the effect of combustible or explosive mixtures of gas that may represent a threat to safety-related equipment if the combustible mixtures of gas are released and ignited. Therefore, this open issue is closed.

3.2.24 113, 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 EPRI 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 3.3.3.4.2 of Appendix B to Chapter 1 of the passive Requirements Document, EPRI recognizes that the ALWR will not make as extensive use of LBHSs as do 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." The elements of resolution only commit generally to develop dynamic tests for LBHSs.

Staff Evaluation: The DSER for Appendix B to Chapter 1, the staff identified Issue 113 as an open issue because it had not yet been generically resolved and had a high safety priority. Issue 113 was resolved in August 1992 with no additional requirements for existing plants. However, in instances where snubbers are required, EPRI has stated that the plant designer will specify use of hydraulic snubbers and their test requirements. The test requirements

will include the method and frequency of testing and will ensure that the dynamic characteristics of the snubber are tested. The staff will evaluate these test requirements to determine their acceptability with respect to NRC requirements in effect at that time during its review of an individual application for FDA/DC. Therefore, this open issue is closed.

3.2.25 121, Hydrogen Control in Large Dry Containments

Issue: As a result of the TMI-2 accident, the Commission promulgated regulatory requirements on hydrogen control in 10 CFR 50.34, "Contents of Applications: Technical Information," and 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Power Reactors." Section 50.34(f) of 10 CFR 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 post-accident 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 passive plants.

EPRI Proposal: In the passive Requirements Document, EPRI proposes that the passive plant design have the capability to ensure that necessary accident prevention and mitigation functions could be performed during and after events in which hydrogen is produced. The passive 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 did not exceed 13 percent under dry conditions.

In Revisions 4 and 5 to the passive Requirements Document, Sections 2.4 and 6.5 of Chapter 5 were modified as follows:

- Under Top-Level Requirements for Accident Mitigation, Section 2.4.2.7: "The design shall ensure that containment integrity is maintained during an accident that releases hydrogen from a 100 percent metal-water reaction of the fuel cladding accompanied by hydrogen burning or the added pressure from post-accident inerting."

- Under Passive Autocatalytic Recombiner System Performance Requirements, Section 6.5.3.1: For passive PWRs, a non-safety-grade passive autocatalytic recombiner (PAR) system will be provided that will "ensure margin in the design for natural circulation mixing of the containment atmosphere by ensuring that local hydrogen concentrations, except at the location of release, will not exceed 10 percent" and "the PAR system shall utilize components with demonstrated capability to combine hydrogen and oxygen under severe accident conditions at a sufficient rate and in a manner compatible with the passive ALWR concept."

Staff Evaluation: The staff's evaluation of this issue is given in this appendix in Section 2.5.3 and under Item I.G of Annex A.

Issue 121 was generically resolved in March 1992, with no additional requirements.

3.2.26 128, 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 TS 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 an emergency diesel generator 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 passive 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 of Appendix B to Chapter 1 lists the following design criteria to address this issue:

- Section 2.2.8 of Chapter 5 requires that each of the redundant components of the engineered safety systems requiring electric power be provided with a appropriately independent Class 1E onsite source of dc power.

- Section 2.3.1 of Chapter 5 requires that the specified safety system functions of engineered safety systems be ensured by providing appropriate redundancy of components and features. The redundancy is necessary to meet the single-failure criterion, (e.g., the most limiting single failure in addition to those failures that by assumption, constitute the accident).

- Section 2.3.2 of Chapter 5 requires that the redundant components and features 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 passive decay heat removal system to be safety grade.

- Section 2.3.6 of Chapter 11 requires each division of engineered safety systems requiring electric power to be provided with physically separated

and electrically independent batteries sized to supply emergency power to the engineered safety system in the event of a loss of all other power sources.

- 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 completely separate power supplies from safety-related loads.

- 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 7.3.2.5 of Chapter 11 requires that each reactor protection and safety systems actuation channel be normally powered from a dedicated Class 1E source that is normally independent of other dc sources. This requirement does not apply to designs with N-2 safety divisions, since, in this case, N-1 divisions are expected to remain available even in the event of a failure of a battery powering a reactor protection channel and a division of safety systems simultaneously.

Staff Evaluation: The design criteria EPRI proposes for the passive 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.

Issue 128 was generically resolved in September 1991, with additional requirements given in GL 91-06 and 91-11.

Some concerns of Issue 128 that involve operational aspects need to be addressed by implementation of appropriate procedures and controls (e.g., through TS). Since the passive 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 or for a COL.

3.2.27 130, 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 while Unit 2 was not yet operational. While Unit 2 was under construction, it was necessary to make a third service water pump available to Unit 1 via a cross-tie 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 cross-tie 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 cross-tie

between plants. Efforts to resolve this issue are to include a survey of Babcock & Wilcox and Combustion Engineering plants to determine if similar multiplant configurations with two ESW pumps per plant and cross-tie 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 this issue are included in Section 3.8.1.4 of Appendix B to Chapter 1 of the passive Requirements Document.

Staff Evaluation: EPRI has concluded in Section 3.8.1.5 of Appendix B to Chapter 1 of the passive Requirements Document that the service water system (SWS) reliability problems can be avoided by not permitting specific design features relative to multiplant cross-ties and not requiring safety-grade service water. The final design of passive plant heat sinks and water flow systems will avoid or minimize, as achievable, the problems described in this generic issue.

Section 3.8.1.4 of Appendix B to Chapter 1 of the passive Requirements Document states that Section 6.4.1 of Chapter 1 requires that, for multiple unit plants on a single site, each unit share only auxiliary support systems such as sewer, auxiliary steam, or security. Therefore, it appears that the SWS pumps will not be shared and the SWS for each unit will be designed to the same requirements as those for a single unit. However, the staff finds that Section 6.4.1 of Chapter 1 does not explicitly preclude shared SWSs. Section 6.4.1 of Chapter 1 states that an analysis will be made of the effect of any failure or testing in shared systems on the maintenance, ALARA, availability, safety, or operability of other systems.

Section 1.2.1.1 of Chapter 5 of the passive Requirements Document states that passive systems (1) are essentially self-contained or self-supported, (2) do not require pumped cooling water to perform their safety function, and (3) for licensing-design-basis events, should be able to perform their safety function independent of operator action or offsite support for 72 hours after the initiating event. However, the staff's review of the above section reveals that support systems are credited for keeping passive systems in a state of readiness at all times by using active components such as ac-powered pumps and valves. Therefore, the staff finds that the SWS may be needed to keep passive systems functional so that they are operable in the event of a design-basis accident.

Section 2.2.14 of Chapter 8 of the passive Requirements Document states that the plant cooling water systems, including the SWS, will be designed with sufficient redundancy of components to meet reliability and availability requirements and that the plant designer may modify the design for these considerations including plant maintainability. EPRI states that Figure 8.5-1 of Section 5.3 of Chapter 8 of the passive Requirements Document shows a single system for simplicity.

Since Issue 130 had not yet been generically resolved by the staff and had a high safety priority, resolution of Issue 130 was identified as an open issue in the DSER for Appendix B to Chapter 1. Issue 130 was generically resolved in September 1991 with additional requirements as given in GL 91-13. The staff will evaluate the specific design during its review of an individual

application for FDA/DC to ensure that any designers proposing shared SWSs at multiplant sites have address GL 91-13. Therefore, this open issue is closed.

3.2.28 135, Steam Generator Overfill

Issue: Steam generator overfill and its consequences have received staff and industry attention because of the frequency and severity of overfill events. Over the years, a number of issues have been raised concerning steam generator overfill, including Issue 66, "Steam Generator Requirements," and Issue 67, "Steam Generator Staff Actions." In order to provide an integrated work plan for the resolution of these issues, Issue 135 was initiated and assigned a medium priority ranking based on the separate evaluation of Issue 67.7.0, "Improved Eddy Current Tests." Resolution of Issue 135 was expected to provide a better understanding of steam generator and secondary steam integrity, including the effects of water hammer on secondary system components and piping as well as the resultant radiological consequences.

EPRI Proposal: EPRI discussed this issue in Section 3.3.4 of Appendix B to Chapter 1 of the passive Requirements Document. EPRI concluded that total resolution of this issue was not applicable to design and thus was not applicable to the ALWR Utility Requirements Document.

Staff Evaluation: This issue was given a medium priority ranking and pursued by the staff. The staff concluded that SGTR and steamline overfill events pose a relatively low public risk, as previously indicated in NUREG-0844 and NUREG-1150. This issue was resolved generically in March 1991 with no new requirements. Therefore, EPRI's position that this issue is outside the scope of the Requirements Document is acceptable.

3.2.29 HF 4.4, Procedures - Guidelines for Upgrading Other Procedures

Issue: This issue addresses the staff's need to develop technical guidance for industry use in upgrading normal operating procedures 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 of Appendix B to Chapter 1 of the passive Requirements Document, EPRI originally stated that Issue HF 4.4 was determined, through the NUREG-1197 screening process, not to be technically relevant to the ALWR design. Subsequently, EPRI addressed this issue in Section 3.10.3 of Appendix B to Chapter 1.

Staff Evaluation: The staff has generically resolved Issue HF 4.4 with no new requirements for existing or future plants.

The staff reviewed Sections 3.4.2, 3.1.3.3.3, 3.1.3.5.1, and 4.1.6.4 of Revision 1 to Chapter 10 of the passive Requirements Document and concludes that all procedures, including operations, maintenance, and test and surveillance procedures, are required by the passive Requirements Document to be developed in accordance with valid human factors principles. Section 3.4.2 of Chapter 10 requires that the main control room operator workstations and other workstations have electronically displayed procedures, where practicable, and that an active simulator be used to validate them. Section 3.4.2.2.2 of

Chapter 10 provides requirements to be followed in practices for these electronically displayed procedures, except where active simulation shows that another practice is preferable. Section 3.4.2.3 of Chapter 10 specifies that hard-copy procedures will be supplied by the M-MIS designer to supplement all electronically displayed procedures and for those control stations for which electronically displayed procedures are not practicable.

However, as stated in the DSER for Appendix B to Chapter 1, the information in these sections of the passive Requirements Document was not sufficient to determine if the requirements for the design, development, and validation of plant procedures were acceptable. 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. Further, EPRI should evaluate issues concerning the use of mixed types of procedures from one control station to the next, as well as the requirement that the active simulator be used for the validation of procedures.

The issue was discussed at a November 3, 1992, meeting between the staff and EPRI. As a result, EPRI revised the passive Requirements Document to state that the specific design features (e.g., electronically displayed procedures) described in the document are to be used simply as a starting point for the design. The staff's evaluation of this issue is contained in Appendix B to Chapter 10 of this report.

3.2.30 HF 5.1, Local Control Stations

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 guidance should be developed to provide (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 will result in the issuance of information (HF 5.1) and review guidance (HF 5.2).

EPRI Proposal: In Section 3.10.3 of Appendix B to Chapter 1 of the passive Requirements Document (Revision 1), 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 passive Requirements Document to improve the M-MI. EPRI also states a number of design features are required in the passive Requirements Document that address these human factors issues, including:

- Section 8.2.1.2 of Chapter 1 (Rev. 0) requires that human factors design principles be consistently applied throughout the design process for each operation work space to reduce operation errors during all plant modes.

- Section 11.1.3.1 of Chapter 1 (Rev. 0) 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 (Rev. 0) 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 (Rev. 0) requires that the interdisciplinary design review group include one member knowledgeable in the principles of human factors.

- Section 2.1.1 of Chapter 10 (Rev. 0) requires that operator limitations not be challenged.

- Section 2.2.10 of Chapter 10 (Rev. 0) requires the application of advanced technology to the main control room to enhance the design.

- Section 4 of Chapter 10 (Rev. 0) requires the application of human factors engineering principles to all control stations.

- Section 4.3 of Chapter 10 (Rev. 0) requires annunciator alarms to be addressed.

- Sections 3.4.5 and 5.6 of Chapter 10 (Rev. 0) require operator aids to be addressed.

- Section 3 of Chapter 10 (Rev. 0) 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 (Rev. 0) requires that the M-MI system designer develop and verify human factor practices for advanced M-MIS 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

The staff has reviewed Section 3.10.3 of Revision 2 of Appendix B to Chapter 1 of the passive Requirements Document in which EPRI commits to design local control stations (LCSs) to the same standards as control stations in the main control room. In the DSER for Appendix B to Chapter 1, the staff stated that this commitment was acceptable; however, EPRI should

- Provide a clear definition of LCSs in the passive 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 passive 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 operating procedure verification. Each introduces potential sources of human error that can be expected to increase with the number of panels involved.

In response to the DSER, EPRI revised Section 4.1 of Chapter 10 of the passive Requirements Document to state in part that LCSs ". . . include[s] single component stations where operators perform such tasks as positioning a manually throttling valve to operate a component locally." The staff has reviewed the revision to the passive Requirements Document to include in the definition, single-component interfaces, and finds EPRI's response addresses its concerns and is acceptable. Therefore, this open issue is closed.

In addition, the issue of functional centralization was discussed further at the November 3, 1992, meeting between the staff and EPRI, and as a result, EPRI revised Section 4.9.2.6 of Chapter 10 of the passive Requirements Document to state in part that "the M-MIS designer shall specifically consider centralizing functions rather than distributing them among several local control stations, where distributing them could place an undue burden on communications, crew coordination, or operator workload associated with carrying out the required operations." Furthermore, the rationale given in the passive requirements document for this requirement references NUREG/CR-3696, "Potential Human Factors Deficiencies in the Design of Local Control Stations and Operator Interfaces in Nuclear Power Plants," and NUREG/CR-5572, "An Evaluation of the Effects of Local Control Station Design Configurations on Human Performance and Nuclear Power Plant Risk," and states in part that "it is the intent in ALWRs that the problems with local control stations in present plants will not recur in the ALWR. Local control functions and tasks should be clearly identified as part of the M-MIS design process." The staff has reviewed the revisions to Section 4.9.2.6 of Chapter 10 of the passive Requirements Document with respect to functional centralization and finds EPRI's response addresses its concerns and is, therefore, acceptable.

(2) Annunciators

EPRI has conducted a considerable amount of research on the improvement of plant alarm systems and has issued several documents providing guidance. Section 4.3 of Chapter 10 (Revision 1) of the passive Requirements Document requires the M-MIS designer to apply the guidance in the relevant EPRI reports, in addition to meeting the specific requirements of the passive Requirements Document. In the DSER for Appendix B to Chapter 1, the staff stated that although EPRI treats alarms as complete, integrated systems, the passive Requirements Document should require a systems analysis that explicitly considers the functional interaction of the alarm system with other plant systems to meet plant and personnel performance objectives. The staff concluded that the passive 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 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 test and evaluation
- quantitative measures to assess human-system performance

In response to the DSER and as a result of further discussions at the November 3, 1992, meeting between the staff and EPRI, Section 11.8.5 of Chapter 1 and Sections 3.1.4.4.4, 3.1.4.4.5, and 4.1.3.3 of Chapter 10 of the passive Requirements Document were revised to address these issues. The staff evaluation of each of these issues is contained in Appendix B to Chapter 10 of this report.

Issue HF 5.1 and 5.2 were generically resolved in June 1993 with no additional requirements.

3.2.31 HF 5.2, Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation

See HF 5.1, above.

Table 3B.1 Generic Safety Issues for Passive Requirements Document

| Issue | Title | NUREG-0933 Status | Vol. III, Ch. 1 App. B Section | Remarks |
|---|---|---|---|---|
| I.D.3 | Safety System Status Monitoring | Resolved 9/93 | 3.10.4 | no requirements |
| I.D.5(3) | Improved Control Room Instrumentation Research - On-Line Reactor Surveillance System | Possible resolution identified | Not relevant per Table B.3-1 note | N/A - research item |
| II.H.2 | | | | |
| II.J.4.1 | Revise Deficiency Reporting Requirements | Resolution available | Not relevant per Table B.3-1 note | Commitments to comply with 10 CFR Part 21 & 50.55(e) in Table 3.B-1 of Ch. 1 of passive Requirements Document |
| B-17 | Criteria for Safety-Related Operator Actions | Medium priority | 3.10.2 | |
| B-55 | Improved Reliability of Target Rock Safety/Relief Valves | Medium priority | Not relevant per Table B.3-1 note | N/A; NUREG-1197 Code 3 |
| B-56 | Diesel Reliability | High priority | Not relevant per Table B.3-1 note | N/A; NUREG-1197 Code 4 |
| B-61 | Allowable Emergency Core Cooling System Equipment Outage Periods | Medium priority | Not relevant per Table B.3-1 note | N/A; NUREG-1197 Code 5 |
| B-64 | | | | |

Table 3B.1 (continued)

| Issue | Title | NUREG-0933 Status | Vol. III, Ch. 1 App. B Section | Remarks |
|---|---|---|---|---|
| C-8* | Main Steamline Isolation Valve Leakage Control Systems | Resolved 3/90 | 3.3.1 | no requirements |
| 15 | Radiation Effects on Reactor Vessel Supports | High priority | 3.4.1 | |
| 23 | Reactor Coolant Pump Seal Failures | High priority | Not relevant per Table B.3-1 note | N/A; NUREG-1197 Code 3 |
| 29 | Bolting Degradation or Failure in Nuclear Power Plants | Resolved - 10/91; GL 91-17, NUREG-1339 | 3.1.1 | |
| 57 | Effects of Fire Protection System Actuation on Safety-Related Equipment | Medium priority | 3.9.1 | |
| 70 | Power-Operated Relief Valve and Block Valve Reliability | Resolved - 6/90; GL 90-06, NUREG-1316 | 3.5.4 | |
| 75 | Generic Implications of Anticipated-Transient Without-Scram Events at the Salem Nuclear Plant | Resolved - 5/90; GL 83-28, GL 83-28, IEBs 83-01 and 83-04 | 3.10.1 | |
| 79 | Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown | Resolved 5/92 | 3.4.2 | GL 92-02, no requirements |
| 83 | Control Room Habitability | Resolution identified | 3.5.1 | |
| 84* | Combustion Engineering Power-Operated Relief Valves | Resolved - 6/90 | 3.5.4 | |

See footnotes at end of table.

Table 3B.1 (continued)

| Issue | Title | NUREG-0933 Status | Vol. III, Ch. 1 App. B Section | Remarks |
|---|---|---|---|---|
| 87 | Failure of High-Pressure Coolant Injection Steamline Without Isolation | Resolved - 12/91 | 3.5.5 | |
| 94 | Additional Low-Temperature Overpressure Protection for Light-Water Reactors | Resolved - 6/90; GL 90-06, NUREG-1326 | 3.3.2 | |
| 105 | Interfacing Systems Loss-of-Coolant Accident at BWRs | High priority | 3.5.2 | |
| 106 | Piping and Use of Highly Combustible Gases in Vital Areas | Medium priority | Not relevant per Table 3.B-1 note | N/A; NUREG-1197 Code 4 |
| 113 | Dynamic Qualification Testing of Large-Bore Hydraulic Snubbers | Resolved 8/92 | 3.3.3 | no requirements |
| 121 | Hydrogen Control for Large, Dry PWR Containments | Resolved 3/92 | 3.5.3 | no requirements |
| 128 | Electrical Power Reliability | High priority | 3.11.1 | |
| 130 | Essential Service Water Pump | Resolved 9/91 | 3.8.1 | GL 91-13 |
| 135 | Failures at Multiplant Sites | | | |
| HF 4.4 | Procedures - Guidelines for Upgrading Other Procedures | Resolved 7/93 | 3.10.3 | no requirements |
| HF 5.1 | Local Control Stations | Resolved 6/93 | 3.10.3 | no requirements |

See footnotes at end of table.

Table 3B.1 (continued)

| Issue | Title | NUREG-0933 Status | Vol. III, Ch. 1 App. B Section | Remarks |
|---|---|---|---|---|
| HF 5.2 | Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation | Resolved 6/93 | 3.10.3 | no requirements |

*Issue not included in Appendix B to NUREG-0933 but addressed in the passive Requirements Document.

NOTES:
N/A - not applicable
Code 3 - applicable only to reactor types and design features not included in current standardized designs
Code 4 - applicable only to operating plants
Code 5 - does not affect plant design (plant operations, management, and procedures; accident management and emergency planning; operator training, qualifications; inspection and maintenance; and operating experience reporting)

## 4 COMPLIANCE WITH INTENT ITEMS

Many of NRC's regulations and some of the guidance in the regulatory guides and the SRP address safety-related features that are present in current LWR design but are not used or identified as safety related in the EPRI passive Requirements Document. As a result, Section 4 of Appendix B to Chapter 1 of the passive Requirements Document states that the application of some of the current NRC regulations and guidance may require interpretations different from those applied to conventional designs because of the difference in the design approach for the passive ALWR. EPRI states that its design criteria generally address the intent of these regulations and guidance.

Section 1 of Appendix B to Chapter 1 contains a list of NRC regulations and guidance that EPRI believes are currently applicable to LWR design and identifies its position with respect to each. In that section, EPRI identifies those items for which it believes the ALWR design should comply with the intent of the regulatory item even though it does not strictly meet the item. Section 4 of Appendix B to Chapter 1 provides an explanation of how that is accomplished.

The staff's review of the entire passive Requirements Document, as discussed throughout this report, provides a clearer picture of the extent to which the document complies with the Commission's regulations. Therefore, the staff does not address these items in this section of the report, but, rather, throughout the report. Table 4B.1 provides a cross reference between the items addressed in Section 4 of Appendix B to Chapter 1 and the lead chapter in the passive Requirements Document where the item is discussed.

Table 4.B-1 Cross Reference Between Items Addressed in Section 4 of Appendix B to Chapter 1 and the Lead Chapter in Passive Requirements Document Where Item Is Discussed

| Item Number | Lead Chapter | Title of Comply With Intent Item |
|---|---|---|
| 4.1 | 3 | CFR 50.34(f)(2)(xiii); Power Supplies for Natural Circulation for PWRs |
| 4.2 | 9 | CFR 50.34(f)(2)(xxviii); Control Room Habitability |
| 4.3 | 5 | CFR 50.62; Requirements for Reduction of Risk From Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants |
| 4.4 | 5 | CFR 50.63; Loss of All Alternating Current |
| 4.5 | 11 | GDC 17; Electric Power Systems |
| 4.6 | 11 | GDC 18; Inspection and Testing of Electric Power Systems |
| 4.7 | 5 | GDC 34; Residual Heat Removal |
| 4.8 | 5 | GDC 35; Emergency Core Cooling |
| 4.9 | 5 | GDC 38; Containment Heat Removal |
| 4.10 | 8 | GDC 44; Cooling Water |
| 4.11 | 7 | GDC 61; Fuel Storage and Handling and Radioactivity Control |
| 4.12 | 11 | Regulatory Guide 1.6; Independence Between Redundant Standby (On-site) Power Sources and Between Their Distribution Systems |
| 4.13 | 11 | Regulatory Guide 1.9; Selection, Design, and Qualification of Diesel-Generator Units Used as Standby (On-site) Electric Power Systems in Nuclear Power Plants |
| 4.14 | 7 | Regulatory Guide 1.13; Spent Fuel Storage Facility Design Basis |

Table 4B.1 (continued)

| Item Number | Lead Chapter | Title of Comply With Intent Item |
|---|---|---|
| 4.15 | 8 | Regulatory Guide 1.27; Ultimate Heat Sink |
| 4.16 | 10 | Regulatory Guide 1.30; Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment |
| 4.17 | 11 | Regulatory Guide 1.32; Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants |
| 4.18 | 11 | Regulatory Guide 1.41; Preoperational Testing of Redundant On-site Electric Power Systems To Verify Proper Load Group Assignments |
| 4.19 | 1 | Regulatory Guide 1.70; Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants |
| 4.20 | 5 | Regulatory Guide 1.75; Physical Independence of Electric Systems |
| 4.21 | 5 | Regulatory Guide 1.79; Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors |
| 4.22 | 11 | Regulatory Guide 1.93; Availability of Electrical Power Sources |
| 4.23 | 11 | Regulatory Guide 1.108; Periodic Testing of Diesel Generator Units Used as On-site Electric Power Systems at Nuclear Power Plants |
| 4.24 | 9 | Regulatory Guide 1.137; Fuel Oil Systems for Standby Diesel Generators |
| 4.25 | 5 | Regulatory Guide 1.155; Station Blackout |
| 4.26 | 1 | Standard Review Plan 2.4.7; Ice Effects |
| 4.27 | 6 | Standard Review Plan 2.4.8; Cooling Water Canals and Reservoirs |

Table 4B.1 (continued)

| Item Number | Lead Chapter | Title of Comply With Intent Item |
|---|---|---|
| 4.28 | 6 | Standard Review Plan 2.4.9; Channel Diversions |
| 4.29 | 8 | Standard Review Plan 2.4.11; Cooling Water Supply |
| 4.30 | 1 | Standard Review Plan 2.4.14; Technical Specifications and Emergency Operation Requirements |
| 4.31 | 1 | Standard Review Plan 3.2.2; System Quality Group Classification |
| 4.32 | 6 | Standard Review Plan 3.5.2; Structures, Systems, and Components To Be Protected From Externally Generated Missiles |
| 4.33 | 5 | Standard Review Plan 5.4.7; Residual Heat Removal (RHR) System |
| 4.34 | 5 | Standard Review Plan 6.2.2; Containment Heat Removal Systems |
| 4.35 | 5 | Standard Review Plan 6.3; Emergency Core Cooling System |
| 4.36 | 5 | Standard Review Plan 6.4; Control Room Habitability |
| 4.37 | 10 | Standard Review Plan 7.3; Engineered Safety Feature Systems |
| 4.38 | 10 | Standard Review Plan 7.4; Safe Shutdown Systems |
| 4.39 | 11 | Standard Review Plan 8.1.3; AC Power System (On Site) |
| 4.40 | 8 | Standard Review Plan 9.1.3; Spent Fuel Pool Cooling and Cleanup System |
| 4.41 | 8 | Standard Review Plan 9.2.1; Station Service Water System |
| 4.42 | 8 | Standard Review Plan 9.2.2; Reactor Auxiliary Cooling Water Systems |

Table 4B.1 (continued)

| Item Number | Lead Chapter | Title of Comply With Intent Item |
|---|---|---|
| 4.43 | 5 | Standard Review Plan 9.2.5; Ultimate Heat Sink |
| 4.44 | 5 | Standard Review Plan 9.3.5; Standby Liquid Control System (BWR) |
| 4.45 | 9 | Standard Review Plan 9.4.1; Control Room Area Ventilation System |
| 4.46 | 9 | Standard Review Plan 9.4.2; Spent Fuel Pool Area Ventilation System |
| 4.47 | 9 | Standard Review Plan 9.4.3; Auxiliary and Radwaste Area Ventilation System |
| 4.48 | 5 | Standard Review Plan 9.4.5; Engineered Safety Feature Ventilation System |
| 4.49 | 9 | Standard Review Plan 9.5.4; Emergency Diesel Engine Fuel Oil Storage and Transfer System |
| 4.50 | 8 | Standard Review Plan 9.5.5; Emergency Diesel Engine Cooling Water System |
| 4.51 | 11 | Standard Review Plan 9.5.6; Emergency Diesel Engine Starting System |
| 4.52 | 11 | Standard Review Plan 9.5.7; Emergency Diesel Lubrication System |
| 4.53 | 11 | Standard Review Plan 9.5.8; Emergency Diesel Engine Combustion Air Intake and Exhaust System |
| 4.54 | 1 | Standard Review Plan 14.2; Initial Test Program - Final Safety Analysis Report |
| 4.55 | 1 | Standard Review Plan 15.1.5; Steam System Piping Failures Inside and Outside of Containment (PWR) |

Table 4B.1 (continued)

| Item Number | Lead Chapter | Title of Comply With Intent Item |
|---|---|---|
| 4.56 | 1 | Standard Review Plan 15.2.6; Loss of Non-emergency AC Power to the Station Auxiliaries |
| 4.57 | 1 | Standard Review Plan 15.2.7; Loss of Normal Feedwater Flow |
| 4.58 | 1 | Standard Review Plan 15.2.8; Feedwater System Breaks Inside and Outside Containment (PWR) |
| 4.59 | 1 | Standard Review Plan 15.5.1 and 15.5.2; Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory |
| 4.60 | 1 | Standard Review Plan 15.6.5; Loss of Coolant Accident Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary |
| 4.61 | 1 | Branch Technical Position CSB 6-1; Minimum Containment Pressure Model for PWR ECCS Performance Evaluation (SRP 6.2.1.5) |
| 4.62 | 11 | Branch Technical Position ICSB PSB-8; Use of Diesel-Generator Sets for Peaking |
| 4.63 | 11 | Branch Technical Position PSB-1; Adequacy of Station Electric Distribution Voltages (SRP 8, Appendix A) |
| 4.64 | 10 | Branch Technical Position PSB-2; Criteria for Alarms and Indications Associated With Diesel-Generator Unit By-passed and Inoperable Status |
| 4.65 | 5 | Branch Technical Position RSB 5-1; Design Requirements of the Residual Heat Removal System (SRP 5.4.7) |
| 4.66 | 1 | NUREG-0737; Clarification of TMI Action Plan |
| 4.67 | 1 | NUREG-0737, Supplement 1; Clarification of TMI Action Plan Requirements |

Table 4B.1 (continued)

| Item Number | Lead Chapter | Title of Comply With Intent Item |
|---|---|---|
| 4.68 | 3 | Standard Review Plan Section 5.4.1.1 and Regulatory Guide 1.14: Pump Flywheel Integrity (PWR) |

# 5 REGULATORY DEPARTURE ANALYSIS

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. In this section, the staff discusses the regulatory departure analyses of those issues identified for the passive 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 SECY-90-016, "Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," issued January 12, 1990, 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, issued March 25, 1991, the staff proposed resolutions for some of these issues for evolutionary designs, which the Commission addressed in its SRMs of June 26, 1990, and August 15, 1991, respectively. The staff also forwarded policy papers to the Commission that address the status of the staff's review of other issues related to the reviews of the ALWR.

The staff prepared two draft Commission papers, "Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements," issued by letter dated February 27, 1992, and "Design Certification Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light-Water Reactor Design," dated June 25, 1992. These draft papers addressed the status of the issues from SECY-90-016 as well as new issues that pertained to both the evolutionary and passive LWR designs. In SECY-93-087, "Policy, Technical and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated April 2, 1993, the staff consolidated the issues from the two draft papers into a single Commission paper.

Forty-two technical and policy issues are discussed in SECY-93-087, which is provided in Annex A of this Appendix, and final staff positions were identified for approximately half of them. The Commission provided guidance on the issue addressed in SECY-93-087 in its SRM dated July 21, 1993, also provided in Annex A. The final staff positions for the remaining issues are in Commission papers addressing either the regulatory treatment of non-safety systems (RTNSS), source term, or emergency planning. The eight policy issues associated with RTNSS are discussed in SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems," dated March 28, 1994, which is provided in Annex B of this appendix. The Commission provided guidance on the issues discussed in SECY-94-084 in its SRM dated June 30, 1994, also provided in Annex B. The staff has determined that the 12 issues associated with the new source term are not policy issues because the Commission, in its SRM associated with SECY-90-016, directed the staff to use the new source term for advanced light-water reactors. The staff's position on the issues associated with source term are presented in the draft

Commission paper, "Source Term Related Technical and Licensing Issues Pertaining to Evolutionary and Passive Light-Water Reactors," dated February 10, 1994, which is provided in Annex C of this appendix. As discussed in Section 2.1.3 of this appendix, EPRI has notified the staff that it is not pursuing the policy issue associated with the simplification of offsite emergency planning at this time. The staff's position on the aspects of emergency planning under 10 CFR Part 52 are provided in Section 2.6 of Chapter 1 of this report and in the draft Commission paper, "Emergency Planning (EP) Under 10 CFR Part 52," dated May 23, 1994, which is provided in Annex D of this appendix.. The staff considers that SECY-93-087 and 94-084 comprise its regulatory analysis of the policy issues identified to date for the passive plant designs.

Certain of the policy issues in SECY-93-087 and SECY-94-084 represent matters specific to the design of these facilities; others address the implementation of the design certification process of 10 CFR Part 52. 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. 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 are discussed for each issue. To aid in identifying its positions, the staff underlined those for which it requested Commission approval.

In the DSER for Appendix B to Chapter 1, the staff stated that since the Commission had not yet received and approved the positions recommended by the staff in the annex of Appendix B, all policy issues discussed in the annex were considered open issues. The Commission has since reviewed the four papers in Annexes A through D and has issued SRMs on the policy issues. All issues considered open in the annex of Appendix B to Chapter 1 have been satisfactorily resolved and the agency positions have been identified.

These issues are considered fundamental to agency decisions on the acceptability of the ALWR designs. For easy identification, Table 5B.1 lists the issues that the DSER identified as applicable to the passive Requirements Document with a cross-reference to the chapters and sections of this report in which they are discussed.

Table 5B.1 Technical and Policy Issues for the Passive Plant Designs

| Issue | Chapter | Section |
|---|---|---|
| Use of physically based source term | 1 | 2.4.1.2 |
| | 1B | 2.5.2; Annex A, Item I.A |
| | 5 | 2,6 |
| Anticipated transients without scram | 1B | 3.2.16; Annex A, Item I.B |
| Mid-loop operation | 1B | Annex A, Item I.C |
| | 3 | 9 |
| Station blackout | 1B | Annex A, Item I.D; Annex B, Item F |
| | 11 | 2.1, 3.2.1 |
| Fire protection | 1B | Annex A, Item I.D |
| Intersystem loss-of-coolant accident | 1B | 3.2.22; Annex A, Item I.E |
| | 3 | 9 |
| Hydrogen control | 1 | 2.5.3 |
| | 1B | 3.2.26; Annex A, Item I.F |
| | 5 | 6.6 |
| Core-concrete interaction - ability to cool core debris | 1B | Annex A, Item I.H |
| | 6 | 4.2.1 |
| High-pressure core melt ejection | 1B | Annex A, Item I.I |
| | 5 | 4.4, 5.4 |
| Containment performance | 1B | Annex A, Item I.J |
| | 5 | 7, 8 |
| Dedicated containment vent penetration | 1B | 2.5.4; Annex A, Item I.K |

Table 5B.1 (continued)

| Issue | Chapter | Section |
|---|---|---|
| | 5 | 7 |
| Equipment survivability | 1 | 4.8.2 |
| | 1B | Annex A, Item I.L |
| Elimination of operating-basis earthquake | 1 | 4 |
| | 1B | 2.1.1; 3.3.1; Annex A, Item I.M |
| In-service testing of pumps and valves | 1 | 12 |
| | 1B | Annex A, Item I.N; Annex B, Item H |
| Industry codes and standards | 1B | 2.1.1; Annex A, Item II.A |
| Electrical distribution | 1B | Annex A, Item II.B; Annex B, Item G |
| | 11 | 4, 6 |
| Seismic hazard curves | 1A | 3.3 |
| | 1B | Annex A, Item II.C |
| Leak before break | 1 | 4.5.5 |
| | 1B | Annex A, Item II.D |
| Classification of main steamline of boiling-water reactor | 1B | 2.3.1; 3.2.8; Annex A, Item II.E |
| | 2 | 3.4.1.5 |
| | 3 | 5.3.3, 5.4.1.5 |
| | 5 | 1.2.3 |
| | 13 | 3 |
| Tornado design basis | 1 | 4.5.2.5 |
| | 1B | 2.1.2; Annex A, Item II.F |

Table 5B.1 (continued)

| Issue | Chapter | Section |
|---|---|---|
| Containment bypass | 1B | Annex A, Item II.G |
| | 5 | 6.4, 7.2 |
| Containment leak rate testing | 1 | 2.5.1 |
| | 1B | Annex A, Item II.H |
| | 5 | 2.1.4, 6.3 |
| Postaccident sampling system | 1B | 2.3.2; Annex A, Item II.I |
| | 3 | 7 |
| Level of detail | 1B | Annex A, Item II.J |
| Prototyping | 1B | Annex A, Item II.K |
| Inspections, tests, analyses, and acceptance criteria | 1 | 10 |
| | 1B | Annex A, Item II.M; Annex A, Item II.L |
| Reliability assurance program | 1 | 6 |
| | 1B | Annex A, Item II.M; Annex B, Item E |
| Site-specific probabilistic risk assessments | 1A | All |
| | 1B | Annex A, Item II.N |
| Severe-accident mitigation design alternatives | 1B | Annex A, Item II.O |
| Generic rulemaking related to FDA/DC | 1B | Annex A, Item II.P |
| Regulatory treatment of non-safety systems | 1 | 2.3.1, 4.3.1, 7, 12.2.1, 12.2.3 |
| | 1B | Annex A, Item III.A; Annex B, Item A |
| | 3 | 2.2 |

Table 5B.1 (continued)

| Issue | Chapter | Section |
|---|---|---|
| Definition of passive failure | 1B | Annex A, Item III.B; Annex B, Item B |
| Thermal-hydraulic stability of the simplified boiling water reactor | 1B | Annex A, Item III.C |
| Safe shutdown requirements | 1B | 2.5.6; Annex A, Item III.D; Annex B, Item C |
| | 3 | 9 |
| | 5 | 3.3, 4.3, 5.3 |
| Control room habitability | 1B | 3.2.18; Annex A, Item III.E; Annex B, Item D |
| Radionuclide attenuation | 1B | 2.3.1; 2.5.2; Annex A, Item III.F |
| | 3 | 7 |
| | 5 | 2.1.6, 2.1.7 |
| Simplification of offsite emergency planning | 1B | 2.1.3; Annex A, Item III.G; Annex D |

# ANNEX A

## DOCUMENTS RELATED TO SECY-93-087 "POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS" APRIL 2, 1993

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# POLICY ISSUE
(Notation Vote)

April 2, 1993 SECY-93-087

**FOR:** The Commissioners

**FROM:** James M. Taylor
Executive Director for Operations

**SUBJECT:** POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

**PURPOSE:**

To present the Commission with recommended positions pertaining to evolutionary and passive light-water reactor (LWR) design certification policy issues and to request that the Commission approve the underlined staff positions presented in this paper.

**SUMMARY:**

In Enclosure 1, the Nuclear Regulatory Commission staff discusses 42 technical and policy issues pertaining to either evolutionary LWRs, passive LWRs, or both. The staff previously identified these issues in the draft Commission papers, "Issues Pertaining to Evolutionary and Passive Light-Water Reactors and Their Relationship to Current Regulatory Requirements," dated February 20, 1992, and "Design Certification Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light-Water Reactor Designs," dated June 25, 1992. After considering the Advisory Committee on Reactor Safeguards (ACRS), industry, and vendor comments, the staff has reached a final position on many of the issues and the staff has underlined the positions for which it is requesting the Commission's approval. The staff also discusses other issues,

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which it concludes may be of interest to the Commission. For these issues, the staff will update the Commission, if appropriate, after reaching a final staff position or as warranted by new, substantive information on the issues.

**BACKGROUND:**

The staff has forwarded several policy papers[1] to the Commission proposing resolutions for policy matters and major technical issues concerning both evolutionary and advanced LWR designs. Two draft Commission papers, cited in the above summary, were released to the public after they were forwarded to the Commission. Those papers supported dialogue between the NRC staff, the ACRS, the Electric Power Research Institute (EPRI), the vendors, and other industry representatives which was focused on resolving the policy and technical issues.

The staff has considered the comments received and reached a final position on many of the issues listed in the draft Commission papers. The staff has finalized its position on 20 issues and requests that the Commission approve the positions recommended in this paper. The staff has also determined that 9 issues discussed in the draft Commission papers are not policy issues and the staff does not anticipate any future interaction with the Commission concerning these issues. For these 9 issues and for the 13 issues which the staff will discuss its final position in future Commission papers, the staff discussions are for the Commission's information only.

**DISCUSSION:**

Enclosure 1 discusses the staff's positions and the current regulatory requirement or interpretation, as well as comments received from the ACRS, the industry, and the vendors regarding 42 technical and policy issues pertaining to evolutionary LWR designs, passive LWR designs, or both. Where appropriate, the staff has included a detailed discussion of the basis for its position on each issue. The staff has also underlined the positions for which it is requesting the Commission's approval.

Enclosure 1 is divided into three sections. Section I discusses issues previously identified to the Commission in SECY-90-016, "Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements." Section II discusses other evolutionary and passive LWR design issues and Section III discusses issues which are applicable only to passive LWR designs. Preliminary analysis and recommendations for issues discussed in Sections II and III were previously transmitted to the Commission in draft Commission papers "Issues Pertaining to Evolutionary and Passive

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[1] Enclosure 3 lists the papers that the staff has forwarded to the Commission regarding policy issues identified for evolutionary and passive advanced light-water reactors. The staff references applicable documents throughout this paper.

Light-Water Reactors and Their Relationship to Current Regulatory Requirements," dated February 20, 1992, and "Design Certification and Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light-Water Reactor Designs," dated June 25, 1992.

Enclosure 2 cross-references issues pertaining to evolutionary, and passive LWR designs with the Commission papers in which the staff has addressed each issue. Enclosure 3 lists Commission papers related to ALWR designs.

The staff developed the recommendations in this paper after:

(1) reviewing current operating reactor designs, evolutionary designs, and passive ALWR design information which was available;

(2) considering operating experience;

(3) considering insights from the available results of the probabilistic risk assessments (PRAs) of LWRs and ALWRs;

(4) considering the Commission's guidance on issues resolved for the evolutionary ALWRs;

(5) completing the draft safety evaluation report for the EPRI Utility Requirements Document (URD) for passive ALWR designs;

(6) completing the final safety evaluation report for the EPRI URD for evolutionary ALWR designs; and,

(7) considering EPRI, ACRS, and industry comments on these issues.

The staff concludes that the positions discussed in Enclosure 1 are fundamental to the Agency's decisions on the acceptability of the evolutionary and passive LWR 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.

The staff plans to forward to the Commission and solicit ACRS and industry comments on at least two additional Commission papers which will discuss issues relating to (1) the regulatory treatment of nonsafety systems in passive designs and (2) use of a physically based source term.

CONCLUSIONS:

The staff requests that the Commission approve the recommended positions for issues pertaining to evolutionary LWR designs. Such approval would enable the staff to proceed with the final design approval and the design certification review of GE Nuclear Energy's (GE) Advanced Boiling Water Reactor and Asea Brown Boveri-Combustion Engineering's (ABB-CE) System 80+ LWR designs.

The staff also requests that the Commission approve the proposed positions for issues pertaining to the passive designs. This will enable the staff to proceed more effectively with its review of Westinghouse's AP600 and GE'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 vendors and EPRI regarding these Commission positions should state this fact.

RECOMMENDATIONS:

The staff recommends that the Commission

(1) Approve the positions underlined in Enclosure 1.

(2) Note that the staff is still considering other policy issues and it will seek the Commission's approval of its positions in the future.

James H. Sniezek /for

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.

Commissioners' comments or consent should be provided directly to the Office of the Secretary by COB Monday, April 19, 1993.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT Monday, April 12, 1993, 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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## POLICY ISSUES ANALYSIS AND RECOMMENDATIONS

This enclosure is divided into three sections. Section I discusses issues previously identified to the Commission in SECY-90-016, "Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements." Section II discusses other evolutionary and passive LWR design issues and Section III discusses issues which are applicable only to passive LWR designs. Preliminary analysis and recommendations for issues discussed in Sections II and III were previously transmitted to the Commission in draft Commission papers "Issues Pertaining to Evolutionary and Passive Light-Water Reactors and Their Relationship to Current Regulatory Requirements," dated February 20, 1992, and "Design Certification and Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light-Water Reactor Designs," dated June 25, 1992.

The staff has reviewed comments from industry and the Advisory Committee on Reactor Safeguards (ACRS) regarding the staff positions discussed in SECY-90-016 and the preliminary staff positions discussed in the two draft Commission papers. The staff has evaluated these comments, continued to dialogue with the industry and the ACRS and, where appropriate, revised our discussions to address industry and ACRS comments. The staff requests that the Commission review the discussions contained in this enclosure and approve the underlined staff positions.

### I. SECY-90-016 Issues

#### A. Use of a Physically Based Source Term

This section provides an overview of the current source term status and discusses source term policy issues previously identified in SECY-90-016. A complete source term discussion pertaining to advanced light-water reactor (ALWR) design will be presented in the forthcoming Commission paper on source term.

For approximately 30 years, the Nuclear Regulatory Commission (NRC) staff has been using the reactor accident source term guidelines contained in Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites (March 1962)," to evaluate design-basis-accident (DBA) analyses.

In SECY-90-016, the staff discussed the methodology for determining compliance with the siting requirements of 10 CFR Part 100 using TID-14844. The staff noted that the assumptions in the TID-14844 methodology considered the uncertainties associated with accident sequences and equipment performance and were intended to ensure that future plant sites would provide sufficient safety margins. This methodology has remained essentially unchanged and many of its original assumptions are considered outdated.

In SECY-90-016, the staff recommended that the Commission approve the following approach for evolutionary ALWRs:

1. Ensure that evolutionary designs meet the requirements of 10 CFR Part 100.

2. Consider deviations from 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.

4. Continue to interact with the Electric Power Research Institute (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 staff requirements memorandum (SRM) of June 26, 1990, the Commission approved the staff's approach in determining the source term for the evolutionary designs. The Commission also directed the staff to modify regulations, regulatory practices, and the review process, as appropriate, to reflect information resulting from source term research.

As a result of this guidance, the NRC staff has developed a new source term based on calculations performed using 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," dated December 1990. The proposed new source term is discussed in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" and was issued in July 1992 as a draft report for public comment. NUREG-1465 discusses radionuclide release fractions, release timing, and chemical forms of fission products that would be released into containment based upon a range of core melt accident scenarios, including failure of the reactor vessel and subsequent molten core-concrete interactions.

The new source term document was not intended and does not provide a specific methodology or implementation guidelines for the staff's licensing review of advanced light-water reactors. Needed information includes quantification of parameters related to the removal or reduction of fission products within containment via the use of engineered safety features such as sprays and filters, and passive processes such as aerosol deposition and plateout. The NRC staff is currently working with Sandia National Laboratory to evaluate fission product removal mechanisms within containment and quantify associated releases to the environment.

EPRI has proposed source terms which are based on bounding severe reactor accidents. EPRI provided the staff with technical justification for their new source terms in correspondence dated October 18, 1990, and February 12, 1991, entitled "Licensing Design Basis Source Term Update for the Evolutionary ALWR," and "Passive ALWR Source Term," respectively. The EPRI-proposed source terms are based on single, enveloping values for bounding severe reactor

accident sequences, using release data obtained from (1) the Severe Fuel Damage Tests at the Power Burst Facility, (2) the Loss of Fluid Test (LOFT) source term measurements, and 3) data from the TMI-2 post accident examination.

For the evolutionary designs, the EPRI source terms address 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 terms also consider 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.

The staff compared the draft source terms in NUREG-1465 with EPRI evolutionary and passive LWR source terms specifically focusing on the accident severity selected, the nature of the release phases, and the timing and magnitude of the important nuclides released. The staff concludes that there is, in general, good agreement between the EPRI-proposed source term for the passive plants and the staff's proposed source term. However, the staff noted that the staff values for the magnitude of low volatility radionuclides were about an order of magnitude higher than those proposed by EPRI. These differences appear to be attributed to EPRI's assumption that little or no fission products will be released to the containment atmosphere from ex-vessel debris following core-concrete interaction. According to EPRI, this is due to the ability of the passive ALWR to provide ample coolant to the reactor cavity/ lower drywell prior to or immediately upon vessel penetration. The NRC staff model does not make this assumption.

GE Nuclear Energy (GE) demonstrated in its standard safety analysis report that the Advanced Boiling Water Reactor (ABWR) design will meet the offsite dose reference values set forth in 10 CFR Part 100 using the current TID-14844 source term. The staff has reviewed the ABWR design and performed an independent analysis of the radiological consequences resulting from a postulated DBA and concluded in the draft final safety evaluation that the ABWR design will meet the dose reference values set forth in 10 CFR Part 100.

Asea Brown Boveri-Combustion Engineering, Inc. (ABB-CE) initially proposed the use of the TID-14844 source term and the existing analyses in the ABB-CE System 80+ standard safety analysis report are based on the TID-14844 source term. However, ABB-CE is evaluating the possibility of adopting the NUREG-1465 source term.

In its AP600 design, Westinghouse proposed the same accident source term as proposed by EPRI for passive plants. In its simplified boiling water reactor (SBWR) design, GE proposed the same accident source term as published in NUREG-1465. The staff continues to interact with the passive plant vendors to resolve source-term related issues.

The staff will continue its approach for review of source terms proposed by EPRI and evolutionary and passive plant vendors (as approved in the June 26, 1990, SRM) without waiting for final agency adoption of NUREG-1465. The staff evaluations of ALWR submittals will utilize the current source term research insights regarding fission product release into containment. In determining the effects of removal mechanisms such as sprays, filters, plateout and aerosol deposition, the staff will utilize engineering judgement and best estimates for the applicable parameters.

The staff is considering several source term related design certification issues. The staff previously identified several source term related policy issues involving control room habitability, radionuclide attenuation, and containment bypass in two draft Commission papers. In addition, several new source term related policy issues appear to be developing which may concern (1) dose assessments for ex-vessel releases which result from a severe accident scenario (inclusion of release from core-concrete interaction), (2) assessment of safety-related equipment qualification and (3) assessment of post-accident sampling capabilities and vital area access provisions. The staff's proposed resolution of these issues will be discussed in a separate Commission paper which will discuss source term related issues.

B. Anticipated Transient Without Scram

As discussed in SECY-90-016, the ATWS Rule (10 CFR 50.62) was promulgated to reduce the probability of an anticipated transient without scram (ATWS) 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 manual operation of the standby liquid control system (SLCS), in lieu of automatic operation as required by 10 CFR 50.62, in the event of an ATWS.

In its SRM of June 26, 1990, the Commission approved the staff's position. However, the Commission directed that the staff should accept an applicant's alternative to the diverse scram system, if the applicant can demonstrate that the consequences of an ATWS are acceptable.

The ABWR design includes a number of features that reduce the risks associated with an ATWS event. These features include a diverse scram system with both hydraulic and electric run-in capabilities on the control rods, a SLCS, and a recirculation pump trip capability. In its letter dated October 9, 1991, GE indicated that it will automate the SLCS and they subsequently provided this information in amendment 20 to the SSAR. In addition, the scram discharge volume has been removed from the ABWR design, eliminating some of the potential ATWS problems associated with the older boiling water reactor (BWR) designs.

The ABB-CE System 80+ design includes a control-grade alternate protection system which is separate and diverse from the safety-grade reactor trip system. This system provides an alternate reactor trip signal and an alternate feedwater actuation signal.

The staff has evaluated the GE ABWR and the ABB-CE System 80+ evolutionary LWR designs and concludes that the designs adhere to the Commission's guidance regarding diverse scram systems.

In its letter of December 6, 1991, EPRI stated that 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.

In its requirements documents, EPRI provides design requirements that are consistent with the staff's position on ATWS discussed in SECY-90-016, as modified by the Commission's SRM of June 26, 1990. In its letter of May 5, 1992, EPRI indicated that its approach to resolving the ATWS issue, for both evolutionary and passive designs, is compliance with the ATWS Rule. EPRI has not proposed design requirements beyond those required to meet the rule.

The passive ALWR vendors have indicated that their designs will comply with the requirements document for passive designs, and the staff is evaluating passive designs to ensure compliance with Commission regulations and guidance regarding ATWS. The staff considers this policy issue resolved.

C. Mid-Loop Operation

In SECY-90-016, the staff stated its concern that decay heat removal capability could be unavailable when a pressurized-water reactor (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 must propose design features to ensure high reliability of the shutdown decay heat removal system.

In its letter of April 26, 1990, the ACRS recommended that the staff consider four additional requirements to resolve this issue. In its memorandum of April 27, 1990, the staff indicated that it would ensure that the four ACRS recommendations would be considered during the review of evolutionary PWR designs.

In its SRM of June 26, 1990, the Commission approved the staff's position and endorsed consideration of the four additional ACRS requirements for mid-loop operation. However, the staff remained concerned that the vendors and EPRI have not adequately evaluated the overall question regarding the vulnerability of the ALWRs during shutdown and low-power operation. This issue was discussed with regard to evolutionary designs in a memorandum to the

Commission dated September 5, 1990. The staff requested that ALWR vendors and EPRI assess shutdown and low-power risk, identifying design-specific vulnerabilities and weaknesses and documenting their consideration and incorporation of design features that minimize such vulnerabilities.

In its letter of December 16, 1991, EPRI submitted proposed changes to the requirements document to address this issue for the evolutionary designs. In its letter of May 5, 1992, EPRI stated that the Passive Requirements Document specifies extensive deterministic requirements to address known shutdown risks based on industry experience. EPRI also requires probabilistic and operational shutdown risk evaluations and analysis and has submitted a revision to the requirements documents to address additional requirements resulting from its review of NUREG-1410, "Loss of Vital AC Power and the Residual Heat Removal System During Midloop Operations at Vogtle Unit 2 on March 20, 1990," and NUREG 1449, "Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States."

The EPRI requirements document and evolutionary PWR designers have provided features to address the issue of mid-loop operation. The staff has reviewed the features for the ABB-CE System 80+ design and concludes that this design adequately implements the Commission's guidance on this issue.

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 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 considers this policy issue resolved.

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 on-site) is unavailable. 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. This power source should be consistent with the guidance in RG 1.155, and should be capable of powering at least one complete set of normal shutdown loads. The staff recommended that the Commission approve its position mandating an alternative ac power source for evolutionary ALWRs. In its SRM of June 26, 1990, 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 ALWR vendors have provided for a large-capacity, alternative ac power source (combustion turbine generator) with the capability to power one complete set of normal safe-shutdown loads. The staff concludes that the EPRI proposal that calls for a "combustion turbine unit" which EPRI has concluded would "achieve diversity of power sources and maximize the overall reliability of

the on-site standby ac power supply system" meets the intent of the Commission guidance on this issue. The staff is still in the process of evaluating the ALWR vendor submittals to ensure acceptable implementation of the Commission's guidance on this issue, but does not expect any related policy matters to result from its review.

Because the passive ALWR designs do not rely on active systems for safe shutdown following an event, EPRI and the passive plant designers have indicated that both safety-related diesel generators and an alternate ac power source should not be required. However, the staff believes that the diesel generators included in passive LWR designs may require some regulatory treatment.

The staff is still evaluating this issue for the passive plant designs. The staff's proposed resolution of this issue will be discussed in a separate Commission paper which will discuss the regulatory treatment of nonsafety systems in passive plant designs.

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. The staff proposed to require that 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 ALWR designers 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, 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.

In its letter of April 26, 1990, the ACRS recommended that the staff consider additional matters in its evaluation of the fire protection designs such as redundant train separation. In its response to the ACRS letter dated April 27, 1990, the staff stated that the proposed requirement to consider the effects of smoke, heat, and fire suppressant migration may warrant separate heating, ventilation, and air conditioning (HVAC) systems, but that other options may be available to the designer. In its SRM of June 26, 1990, the Commission approved the staff's position, as supplemented by the staff's response of April 27, 1990.

In a letter dated May 5, 1992, EPRI stated, "The ALWR program requirements have been specified to provide for three-hour fire barriers between redundant safety systems and to prevent the migration of smoke and hot gases between compartments containing redundant safety systems by requiring design of separate HVAC systems to serve redundant trains of safety equipment. Exceptions to these requirements are the main control room and reactor containment."

In its letter dated August 17, 1992, ACRS stated, "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." The ACRS concern with shared HVAC relates to the need for adequate isolation of such systems during certain disruptive events (such as fires, floods, or pipe breaks). ACRS stated:

> 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.

The staff maintains that the proposed requirement to consider the effects of smoke, heat, and fire suppressant migration may warrant separate HVAC systems, but that other options may be available to the designer.

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 guidance on this issue, and does not expect any related policy matters to result from its review.

EPRI has specified requirements for the passive designs similar to 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 has not identified any unique features of the passive 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 position that the passive plants should also be reviewed against the enhanced fire protection criteria approved in the Commission's SRM of June 26, 1990.

F. Intersystem Loss-of-Coolant Accident

In SECY-90-016, the staff recommended that evolutionary ALWR designers should 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 RCS to withstand the full RCS pressure. The staff further recommended that systems that have not been designed to full RCS pressure should include:

- 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; and,
- high-pressure alarms to warn control room operators when rising reactor coolant (RC) pressure approaches the design pressure of attached low-pressure systems and both isolation valves are not closed.

In its SRM of June 26, 1990, the Commission approved the staff's position on intersystem LOCA, provided that all elements of the low-pressure system are considered (including instrument lines, pump seals, heat exchanger tubes, and valve bonnets).

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 guidance on this issue, but does not expect any related policy matters to result from its review.

EPRI has specified requirements for the passive designs similar to those for the evolutionary designs. The passive ALWR vendors have indicated that their designs will comply with the applicable requirements document.

In its letter of May 5, 1992, EPRI stated that in order to ensure that the guidance of SECY-90-016 is applied to all systems and subsystems connecting to the RCS, the Utility Requirements Document (URD) was revised to include the following specific requirements:

- All systems and subsystems connected to the RCS which extended outside the primary containment boundary must be designed to the extent practicable to an ultimate rupture strength (URS) of at least equal to full RCS pressure.
- The designer must determine by evaluation that, for interfacing systems or subsystems which do not meet the full RCS URS requirement, the degree and quality of isolation or reduced severity of the potential pressure challenges are sufficient to preclude an intersystem LOCA.
- Additional testing and control room alarm capabilities must be implemented to help reduce the probability of an intersystem LOCA.

In general, the staff has found that these requirements are consistent with the staff position. However, as stated in the draft safety evaluation report (SER) on the passive URD, it will be necessary for the plant designer to demonstrate that any interfacing system for which the URS is not at least equal to full RCS pressure could not practically be designed to meet such a criterion. The degree of isolation or number of barriers (for example, three isolation valves) is not sufficient justification for using low-pressure components that can practically be designed to the full RCS URS criterion. In addition, piping runs should be designed to meet the full RCS URS criterion, as should all associated elements (such as flanges, connectors, packing, valve stem seals, pump seals, heat exchanger tubes, valve bonnets, and RCS drain and vent lines). The plant designer should make every effort to reduce the level of pressure challenge to all systems and subsystems connected to the RCS.

The staff has not identified any unique features of the passive plant 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 position that the passive plants should also be reviewed for compliance with the intersystem LOCA criteria approved in the Commission's SRM of June 26, 1990.

G. Hydrogen Control

Containments are required to be designed for control of hydrogen generation following an accident and 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. For example, 10 CFR 50.34(f)(2)(ix), *Additional TMI-related requirements*, requires a hydrogen control system that can safely accommodate hydrogen generated by the equivalent of a 100-percent fuel-clad metal water reaction. The system must also ensure 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.

Because of the uncertainties in the phenomenological knowledge of hydrogen generation and combustion, the staff recommended in SECY-90-016 that evolutionary ALWRs should be designed, as a minimum, to satisfy the following criteria:

- accommodate hydrogen generation equivalent to a 100-percent metal-water reaction of the fuel cladding;
- limit containment hydrogen concentration to no greater than 10 percent; and,
- provide containment-wide hydrogen control (such as igniters or inerting) for severe accidents.

The staff recommended that the Commission approve its position that the requirements of 10 CFR 50.34(f)(2)(ix) should remain unchanged for evolutionary ALWRs. In its SRM of June 26, 1990, the Commission approved the staff's position.

In its letter of December 6, 1991, EPRI stated that its 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 containment atmosphere. The ABB-CE System 80+ has a hydrogen mitigation system consisting of igniters to accommodate the hydrogen production from a 100-percent fuel-clad metal-water reaction and maintain the average containment hydrogen concentration below 10 percent. The staff is in the process of evaluating these submittals to ensure acceptable implementation of the Commission's guidance on this issue but does not expect any related policy matters to result from its review.

In a letter dated May 5, 1992, EPRI indicated that the Requirements Document for evolutionary plants had been revised to require use of a containment-wide hydrogen control system that meets the requirements of 10 CFR 50.34(f). In addition, EPRI indicated that changes were being developed to require a hydrogen control system suitable for passive plants, which would also meet the regulations. The ALWR program intends to specify hydrogen control requirements consistent with NRC staff guidance. The passive ALWR vendors have indicated that their designs will comply with the applicable requirements document.

In its letter of August 17, 1992, ACRS indicated that it supported the staff's recommendation that hydrogen mitigation systems accommodate a 100-percent fuel-clad metal-water reaction. ACRS recommended that the staff perform an analysis similar to that conducted in support of the resolution of Generic Issue 106, "Piping and Use of Highly Combustible Gases in Vital Areas" on the impact of hydrogen combustion and possible detonation including stratification, before establishing an average hydrogen concentration limit.

The staff has performed numerous studies and conducted several experimental programs to better understand the behavior of hydrogen combustion and potential concentration gradients within the primary containment. While it is clear that additional analyses would add to the overall data base, the staff believes that a *sufficient base exists today to go forward with licensing criteria*. With respect to the insights gained from Generic Issue 106, the staff has extensively examined the potential for and consequences of detonations. In the last decade, the results of these efforts have been published in NUREG/CR-4905, 4961, 5275, and 5525. The current NRC research program is also examining the generic issue of local explosions and the initiation of detonations by jet flames.

The staff recommends that the Commission approve the position that passive plants should be designed, as a minimum, to the same requirements applied to evolutionary designs, as approved by the Commission's SRM of June 26, 1990. Specifically, passive plant designs must:

- accommodate hydrogen generation equivalent to a 100-percent metal-water reaction of the fuel cladding;
- limit containment hydrogen concentration to no greater than 10 percent; and,
- provide containment-wide hydrogen control (such as igniters or inerting) for severe accidents.

H. Core Debris Coolability

In the unlikely event of a severe accident in which the core melts 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.

In SECY-90-016, the staff recommended that the Commission approve the general criteria that evolutionary ALWR designs:

- provide sufficient reactor cavity floor space to enhance debris spreading; and,
- provide for quenching debris in the reactor cavity.

In its SRM of June 26, 1990, 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. Debris coolability is an area in which there is active ongoing experimental research including relatively large scale testing jointly sponsored by EPRI and NRC. The results of tests were expected to demonstrate the early quenchability of core debris within the reactor cavity. However, these tests were indeterminate in proving quenchability and indicated the need to consider the potential for continued core-concrete interaction. Concrete can be used as a sacrificial barrier, for both the liner and structural components, to accommodate potential longer periods of core-concrete interaction. The staff now 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 proposed floor

sizing criteria of 0.02 $m^2$/MWt, and provisions to flood the lower drywell or reactor cavity. The staff neither supports nor disputes the EPRI floor sizing criteria of 0.02 $m^2$/Mwt. Instead, the staff concludes that it is appropriate to review the specific vendor designs to determine how they address the general criteria discussed above (including protecting structural components with concrete) 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 core-concrete interactions 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.

The 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. Because the staff neither supports nor disputes particular floor sizing criteria, vendors should ensure that the containment can withstand the environmental conditions (pressure and temperature) and structural challenge caused by core-concrete interactions. For the range of severe accidents of concern, vendors should realistically estimate the amount of core-concrete interaction that will occur, and ensure that the containment will accommodate the resultant conditions for approximately 24 hours without loss of containment integrity. Where insufficient data exist to develop realistic estimates, vendors 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 available to interact with the concrete.

The staff concludes that incorporation of mitigative measures (to the extent practical) and assurance of containment integrity for approximately 24 hours will provide defense-in-depth as well as an appropriate degree of robustness in the containment design.

In its letter of May 5, 1992, EPRI indicated that the requirements documents specify requirements to address debris coolability including cavity/lower drywell area to permit spreading, cavity/lower drywell flooding to quench debris, and protection of the containment boundary from debris attack. The requirements documents also specify that containment loads from dominant core damage sequences be evaluated. The passive ALWR vendors have indicated that their designs will comply with the applicable EPRI requirements document.

The staff recommends that the Commission approve the position that both the evolutionary and passive LWR designs meet the following criteria:

- provide reactor cavity floor space to enhance debris spreading;
- provide a means to flood the reactor cavity to assist in the cooling process;

- protect the containment liner and other structural members with concrete, if necessary; and,

- ensure that the best estimate environmental conditions (pressure and temperature) resulting from core-concrete interactions do not exceed Service Level C for steel containments or Factored Load Category for concrete containments, for approximately 24 hours. Ensure that the containment capability has margin to accommodate uncertainties in the environmental conditions from core-concrete interactions.

I. High-Pressure Core Melt Ejection

In SECY-90-016, the staff recommended that the Commission approve the position that evolutionary ALWR designs should include a depressurization system and cavity design features to contain ejected core debris in order to reduce the potential for containment failure as a result of direct containment heating (DCH). The staff is concerned that this event might result from the ejection of molten core debris under high-pressure from the reactor vessel. Such an ejection might result in wide dispersal of core debris, rapid oxidation, and extremely rapid addition of energy to the containment atmosphere.

In its SRM of June 26, 1990, the Commission approved the staff's position. The Commission also directed 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 (1) ledges or walls that would deflect core debris and (2) an indirect path from the lower reactor cavity to the upper containment. The staff will review the LWR designs relative to the above criteria.

In its letter of May 5, 1992, EPRI indicated that the requirements document specifies an RCS depressurization system and cavity retention capability for both evolutionary and passive plants. EPRI further indicated that since the passive plant emergency core cooling system (ECCS) relies on RCS depressurization, redundancy and diversity have been specified for the depressurization system to ensure very high reliability.

In its letter of August 17, 1992, ACRS indicated that because direct containment heating is an extremely improbable event, two modes of coping with the possibility are not needed. ACRS stated that because of the possible safety benefits for other events, reliable depressurization is the preferred approach.

The staff agrees with the ACRS assertion that a reliable depressurization system is needed. However, the staff proposes to provide a design concept with a degree of consequence mitigation along with a certain amount of accident prevention. The depressurization system retains a degree of uncertainty. Such questions as the rate of depressurization, the timing for

operator initiation of manual depressurization, and the cut-off pressure may never be totally resolved. As a result, the staff believes that a design can be developed to decrease the direct flight path to the upper containment at little or no added expense.

The plant designers have provided features to address this issue for evolutionary ALWR designs. The staff is in the process of evaluating their submittals to ensure acceptable implementation of the Commission's guidance on this issue. The staff's preliminary review of the passive ALWRs has also identified the importance of RCS depressurization to the safe shutdown of the plant during transients or accidents. RCS depressurization 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 should include a highly reliable depressurization system.

The staff recommends that the Commission approve the general criteria that the evolutionary and passive LWR designs

- provide a reliable depressurization system; and,
- 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 position to evaluate evolutionary ALWRs using a conditional containment failure probability (CCFP) of 0.1, or a deterministic containment performance goal that offers comparable protection. The staff concluded that the following general criterion would be an appropriate substitute for a CCFP in evaluating evolutionary ALWR containment performance during a severe-accident challenge:

> 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 staff proposed this containment performance goal to ensure that the containment will perform its function in the face of most credible severe accident challenges.

In its SRM of June 26, 1990, the Commission approved the use of a 0.1 CCFP as a basis for establishing regulatory guidance for the evolutionary ALWRs. The Commission directed, however, that this objective should not be imposed as a requirement, and that the use of the CCFP should not discourage accident

prevention. The Commission also directed the staff to review and submit to the Commission suitable alternative, deterministically established containment performance objectives providing comparable mitigation capability that may be considered by the applicants.

In this paper and in SECY-90-016, the staff has identified the major challenges to the containment (such as hydrogen burns or core debris interactions with water and containment structures) and the need to provide the means for mitigation of these challenges. Nonetheless, the containment performance goal acts as defense-in-depth 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 interaction between molten fuel and coolant, and the resulting steam and hydrogen generation (and any dynamic forces due to ex-vessel fuel-coolant interactions outside the vessel) on the integrity of the containment, consistent with the containment performance goal. The evaluation of containment bypass sequences will be addressed on a vendor-specific basis during the staff's review of ALWR designs.

The intent of both the CCFP and the alternative deterministic performance criteria discussed above is to provide a 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.

In its letter of August 17, 1992, ACRS stated that the staff has not developed an adequate technical position relating to requirements for containment performance in passive designs. ACRS also contends that the CCFP approach should be used in developing regulatory requirements and not merely passed on to the applicants. With regard to deterministic evaluation of containment performance, ACRS implied that the staff should identify the containment challenges, or how they are to be quantified, on a generic basis. ACRS endorsed the criteria recommended in its letter of May 17, 1992, "Proposed Criteria to Accommodate Severe Accidents in Containment Design."

During the evolutionary ALWR reviews, the staff conducted a thorough review to ensure that a probabilistic CCFP would not be used in a way that could detract from a balanced approach of severe accident prevention and consequence mitigation. The ABWR review is nearly complete and, at this time, the staff believes GE has met the CCFP goal of 0.1 for internal events. The ABB-CE System 80+ design is also expected to meet the CCFP goal of 0.1.

Through the review of the evolutionary ALWR designs, the staff recognized the limitations of a CCFP approach. Specifically, as accident prevention is increased by decreasing core damage frequency, the ability of the containment to withstand events of even lower probability becomes less clear. The limitations of a CCFP approach are also evident in the uncertainties prevalent within a probabilistic risk assessment (PRA). Because of these limitations,

vendors have provided some deterministic analysis to complement the CCFP approach by addressing uncertainties in the severe accident phenomena and calculation of the CCFP. In particular, uncertainties surrounding the issue of debris coolability have led the staff to conclude that deterministic best-estimate analyses should complement the CCFP approach. In spite of its limitations, the staff believes that the CCFP approach ensures that evolutionary ALWRs maintain a balance between accident prevention and consequence mitigation.

In its letter of May 5, 1992, EPRI indicated that the requirements document specifies a set of deterministic containment performance requirements supported by PRA. These requirements address a set of severe accident containment challenges and specify that ASME Service Level C limits be met for risk-significant sequences (expected to be low-pressure core melt into an intact containment). EPRI indicated that the proposed staff criteria are generally consistent with the ALWR requirements, except in the challenges to be considered in demonstrating the 24-hour Service Level C goal. EPRI has provided a containment performance study for both evolutionary and passive plant designs that identifies 23 postulated containment challenges and failure modes along with specification of design features for their mitigation and requirements for deterministic analysis. The staff has reviewed the listing of identified challenges and believes that it represents a complete set.

In SECY-92-292, "Advance Notice of Proposed Rulemaking on Severe Accident Plant Performance Criteria for Future ALWRs," dated August 24, 1992, the staff proposed that the Commission issue an Advance Notice of Proposed Rulemaking (ANPR) on Severe Accident Plant Performance Criteria for Future LWRs. Intended to be used for passive LWRs, the ANPR contains three distinct options that could be followed to establish severe accident containment performance requirements. The three options include a hardware-oriented rule, a phenomena-oriented rule, and a general design criteria (GDC)-oriented rule that was outlined in detail in an ACRS letter dated May 17, 1991.

In an SRM dated September 17, 1992, the Commission directed the staff to place the ANPR in the Federal Register for a 90-day comment period. The staff will review the containment performance goal and severe accident guidance following receipt of all public comment and after meeting with Westinghouse and GE representatives to discuss their passive LWR designs. The staff will also discuss these issues with ACRS.

The staff is currently evaluating the containment performance of the individual ALWR designs to ensure that all potential sequences that may be identified during the staff's review are adequately addressed. The staff will evaluate the criteria used by the vendor to determine the challenges to the containment.

In SECY-90-016, the staff indicated that 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. Because both containment performance goals are acceptable and given the limitations inherent within the CCFP approach, the staff believes the deterministic containment performance goal should be pursued for the passive ALWR designs.

Although the staff is still evaluating this issue and expects further insight from public comments on the ANPR, the staff concludes that it is appropriate to proceed with interim severe accident containment performance criteria. These interim criteria are intended to reflect the lessons learned during the review of the evolutionary designs. The staff recommends that the Commission approve the position to use the following deterministic containment performance goal in the evaluation of the passive ALWRs:

> The containment should maintain its role as a reliable, leak-tight barrier (for example, by ensuring that containments stresses do not exceed ASME Service Level C limits for metal containments, or Factored Load Category for concrete containments) approximately 24 hours following the onset of core damage under the more likely severe accident challenges and, following this period, the containment should continue to provide a barrier against the uncontrolled release of fission products.

The staff will inform the Commission if it identifies additional policy issues as a result of (1) reviewing ALWR designs, or (2) evaluating comments on the ANPR concerning "Severe Accident Plant Performance Criteria for Future LWRs."

K. Dedicated Containment Vent Penetration

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 ABWR. This system should be designed to avoid gross containment failure resulting from postulated slow rising overpressure scenarios that could result from postulated multiple safety system failures.

In its SRM of June 26, 1990, the Commission approved the staff's recommended use of the containment overpressure protection system for the ABWR, subject to a comprehensive regulatory review to consider the "downside" risks with the mitigation benefits of the system. In addition, the Commission directed the staff to ensure that the design provides full capability to maintain control over the venting process.

In its letter of May 5, 1992, EPRI indicated that containment venting was an optimization issue in the EPRI requirements documents. According to the ALWR Utility Steering Committee representatives, specific containment overpressure protection for evolutionary ALWR designs, either through the size and strength of containment or through installation of an overpressure protection relief, is considered an acceptable approach.

EPRI indicated that passive plant design features that address the containment overpressure challenge include highly reliable, redundant, and diverse passive safety-grade decay heat removal, automatic depressurization, and containment cooling. EPRI recommended that the NRC not require a containment vent in evolutionary PWR containments and passive ALWRs.

In its letter of August 17, 1992, ACRS indicated that the Commission should make a generic judgement about the acceptability of containment vents for light water reactors. ACRS contends that this should be part of establishing general criteria for containment designs, as proposed in their letter of May 17, 1992.

The staff considers the containment vent as one of many plant systems that can be used to mitigate the consequences of an accident. If acceptable analyses indicate that a vent would not be needed to meet the severe accident criteria, such as the containment performance goal, the staff would not propose to implement a vent requirement.

Because of the current stage of design development and review, the staff has insufficient information at this time to determine whether a containment vent is necessary for passive plant designs. 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 position that the need for a containment vent for the passive plant designs should be evaluated on a design-specific basis.

L. Equipment Survivability

In SECY-90-016, the staff recommended that the Commission approve the position that features provided only for severe-accident protection need not be subject to the environmental qualification requirements of 10 CFR 50.49; quality assurance requirements of 10 CFR Part 50, Appendix B; or redundancy/diversity requirements 10 CFR Part 50, Appendix A. The reason for this judgement is that the staff does not believe that severe core damage accidents should be treated in the same manner traditionally used for design-basis accidents (DBAs) because of significant differences in their likelihood of occurrence. However, SECY-90-016 further stated that mitigation features must be designed to provide 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 provided for DBAs 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 ALWR designs, the staff will evaluate the ALWR vendors identification of the equipment needed to perform mitigative functions and the conditions under which the mitigative systems must operate. Equipment survivability expectations under severe-accident conditions should consider the circumstances of applicable

initiating events (such as station blackout or earthquakes) and the environment (including pressure, temperature, and 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 SRM of June 26, 1990, the Commission approved the staff's position. In its letter of May 6, 1991, the staff clarified its position that these criteria would be applied to features provided only for severe accident mitigation.

The EPRI requirements document and the evolutionary ALWR designers have indicated that their submittals are consistent with these criteria. The staff is in the process of evaluating their submittals to ensure acceptable implementation of the Commission's guidance on this issue, 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 applicable EPRI requirements document. In its letter of August 17, 1992, ACRS agreed with the staff position discussed above.

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 criteria established for evolutionary plants. Therefore, the staff recommends that the Commission approve the position that passive plant design features provided only for severe-accident mitigation need not be subject to the environmental qualification requirements of 10 CFR Section 50.49; quality assurance requirements of 10 CFR Part 50, Appendix B; and redundancy/diversity of requirements 10 CFR Part 50, Appendix A. As discussed in SECY-90-016, the staff concludes that guidance such as that found in Appendices A and B of RG 1.155, "Station Blackout," is appropriate for equipment used to mitigate the consequences of severe accidents.

M. Elimination of Operating-Basis 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 the intensity 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 SRM of June 26, 1990, the Commission approved the staff's recommendation.

In a plant optimization subject, EPRI requested that the staff evaluate the total elimination of the OBE from the design of systems, structures, and components (SSC) in nuclear power plants. In its letter of April 26, 1990, the ACRS also recommended this approach. In evaluating the decoupling of the OBE from the SSE, the NRC staff 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 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 already been implemented to mitigate the situation of having the OBE significantly control the design.

The elimination of the OBE response analysis would require performance of all current OBE design-related checks for a fraction of the SSE. The staff is working with the industry to develop various design alternatives for the ABWR to supplement 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 evaluating earthquake cycles on fatigue and relative seismic anchor motion effects that are based on the OBE. The staff and industry are supplementing the Code with design rules that would account for fatigue and seismic anchor motion effects based on the SSE. In addition, the NRC guidelines for postulating the number and location of pipe ruptures are also derived from the OBE. When the OBE is eliminated from design, these loading calculations may need to be performed using the SSE after establishing appropriate new allowable limits.

The staff's proposed amendment of Appendix A to 10 CFR Part 100 would allow the option to eliminate the OBE from design certification when the OBE is established at less than or equal to one-third the SSE. In this manner, the OBE serves 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 staff assessed the safety margins of several aspects of nuclear plant design when the OBE is eliminated from consideration. The industry and staff recognize that earlier seismic criteria caused the OBE to control certain aspects of the plant design (such as the piping systems). The industry and the staff view the "controlling" nature of the OBE design as an 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 SSC 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. On the basis of analyses, tests, and engineering judgment, the staff has determined that the structural design produced using SSE load combinations envelop the load combinations produced using 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.

For analyses of safety-related structures, the effects of the relative displacements between adjacent structures need to be considered for earthquakes. With the elimination of the OBE, these effects should be considered for the SSE and would also be needed for input into the piping design as discussed later in this issue. In RGs 1.27 and 1.143, the staff recommends that the designs of the ultimate heat sink and radioactive waste structures, respectively, be evaluated only for OBE loading using the appropriate load combinations and limits for the OBE. When the OBE is eliminated from the ABWR, the staff found that GE proposed to design these buildings and structures to the SSE loadings. The staff concluded that using SSE loadings and the appropriate load combinations and limits provides a bounding design comparable to that provided in the regulatory guides. The staff will also review alternative methods that might be proposed by other vendors on a case-by-case basis.

A 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).

After reviewing these aspects, the staff concludes that, for primary stresses, if the OBE is established at one-third the SSE, the SSE load combinations 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.

Eliminating the OBE will, however, 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 these 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 will incorporate the guidelines into an SRP revision or into a regulatory guide, as necessary.

To account for earthquake cycles in the fatigue analyses of piping systems performed until the new guidance is issued, the staff proposes using two SSE events with 10 maximum stress cycles per event (20 full cycles of the maximum SSE stress range). This is equivalent to the cyclic load basis of one SSE and five OBE events, as currently recommended in SRP Section 3.9.2, when accounting for differences in the structural damping between the OBE and SSE and for a 60-year (instead of a 40-year) plant life. Alternatively, the staff proposes that number of fractional vibratory cycles equivalent to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Appendix D of IEEE Standard 344-1987.

The ASME Boiler and Pressure Vessel Code (ASME Code), Section III, Paragraph NC/ND-3655 specifies that seismic anchor displacement effects need not be considered for Service Level D. However, the ASME Code requires that seismic anchor motion stresses be considered for Service Level B, for which the OBE has traditionally been the designated seismic loading. If the OBE were 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 proposes that the effects of anchor displacements in the piping caused by an SSE be considered with the Service Level D limit. The staff's recommendation will correct this anomaly and will require an evaluation of seismic anchor motion effects for the SSE together with the effects of normal conditions as required by 10 CFR Part 50, Appendix A, GDC 2. Their effects would be evaluated to a Service Level D limit for which the SSE has traditionally been the designated seismic loading.

The staff proposes that existing staff guidelines ensuring the functionality of safety-related components and supports under SSE loading conditions be maintained. For example, when safety-related equipment is qualified by analysis only, the stress limit should remain in the linear-elastic range for SSE loading. Similarly, the function of the supported system must be taken into account. As specified in RG 1.124, "Service Limits and Loading Combinations for Class 1 Linear-Type Components Supports," the ASME Service Level B limits of Subsection NF (or other justifiable limits approved by the staff) should be used to ensure that systems which normally prevent or mitigate consequences of SSE-related events will operate adequately regardless of plant condition.

Pipe rupture is a rare event that can be caused by errors in 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 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 in a piping system designed to an SSE 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 the Commission approve the approach to eliminate the OBE from the design of systems, structures, and components. When the OBE is eliminated from the design, no replacement earthquake loading should be used to establish the postulated pipe rupture and leakage crack locations. The staff recommends that the criteria for postulating pipe ruptures and leakage cracks in high- and moderate-energy piping systems be based on factors attributed to normal and operational transients alone. However, for establishing pipe breaks and leakage cracks due to fatigue effects, calculation of the cumulative usage factor should continue to include seismic cyclic effects.

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 (such as control of erosion/corrosion or use of upgraded piping materials). The guidelines for environmental qualification and compartment pressurization are currently based on the mechanistic break locations. Therefore, the staff proposes that the mechanistic pipe break and high-energy leakage crack locations determined by the piping high stress (without the OBE) and fatigue locations may be used for equipment environmental qualification and compartment pressurization purposes.

Eliminating the OBE from explicit design consideration affects several aspects of the seismic qualification of safety-related mechanical and electrical equipment. When the equipment qualification is performed by analysis, the acceptance criteria are derived from the ASME Code. The effect of eliminating the OBE from equipment qualification by analysis should be negligible. It is well known that mechanical equipment (such as pumps and valves) is, in general, seismically rugged when adequately anchored. It is also known that operability limits for mechanical equipment are generally established through maximum permissible moments and forces or tolerance limits based on available clearances that are controlled by the SSE (rather than the OBE). Therefore, for mechanical equipment, elimination of OBE from qualification by analysis should not reduce any margin. Also, some electrical equipment may be qualified by an analysis requiring demonstration that five OBE events followed by one SSE event do not cause a failure of the equipment to perform its safety function. With the elimination of OBE, analysis checks for fatigue effects can be performed at a fraction of the SSE (such as 50 cycles at one-half of the SSE peak amplitude, or 150 cycles at one-third of the SSE peak amplitude).

When equipment qualification for seismic loadings is performed by analysis, testing, or a combination of both, the staff recommends the use of IEEE Standard 344-1987, as endorsed in RG 1.100, Revision 2. For such analysis, the selection of the service limit level for different loading combinations will ensure the functionality of the equipment during and following a SSE. For testing, IEEE Standard 344-1987 details requirements for performing seismic qualification using five OBE events followed by an SSE event. Where complex mathematical models are based solely on calculated structural parameters, verification testing should be performed.

With the elimination of the OBE, two alternatives exist that will essentially maintain the requirements provided in IEEE Standard 344-1987 to qualify equipment with the equivalent of five OBE events followed by one SSE event (with 10 maximum stress cycles per event). Of these alternatives, the staff concludes that equipment should be qualified with five one-half SSE events followed by one full SSE event. Alternatively, a number of fractional peak cycles equivalent to the maximum peak cycles for five one-half SSE events may be used in accordance with Appendix D of IEEE Standard 344-1987 when followed by one full SSE. The staff will conduct research to verify 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.

Lastly, the design of ALWRs using a single-earthquake (that is, SSE) design is predicated on the adequacy of pre-earthquake planning and post-earthquake damage inspections that are to be implemented by the combined operating license (COL) applicant. The staff proposes that the COL applicant submit to the NRC staff, as a part of its application, the procedures it intends to use for pre-earthquake planning and post-earthquake inspections. The staff is currently developing a regulatory guide for pre-earthquake planning and post-earthquake operator actions.

In its letters dated May 5, 1992, and August 21, 1992, EPRI representatives commented on the staff's positions concerning eliminating the OBE from design as discussed in the two draft Commission papers. EPRI agrees with the staff's recommendation to eliminate the OBE from design. The EPRI requirements documents will require a seismic margins assessment, which demonstrates a margin for an earthquake substantially larger than the SSE. However, EPRI does not fully agree with the number of earthquake cycles to be used in fatigue evaluation and equipment qualification. The EPRI recommended that the NRC guidelines be similar to those contained in paragraph N-1214 of ASME Code, Section III, Appendix N, which provides that no more than two OBE events with a total of 20 full stress cycles (that is, 20 cycles at one-half SSE) be used.

Similarly, in its letter dated September 17, 1992, Westinghouse Electric Corporation representatives commented on the staff's position regarding the elimination of the OBE and noted that the staff's proposed position was overly conservative in the number of earthquake cycles to be considered. Westinghouse proposed to adopt the guideline of 20 cycles at one-half of the SSE response, as specified by the EPRI URD for passive plants.

The staff's evaluation indicates, however, that the EPRI and Westinghouse recommendations provide less stringent design requirements than those currently prescribed by the staff guidelines of SRP Section 3.9.2. Additionally, the current staff guidelines were used for the review of nuclear plants with a 40-year life or less.

The staff positions presented above have been revised since the draft Commission papers to account for both a 60-year plant life and the differences in the structural damping used for the OBE and SSE. (Component damping was not specified because EPRI pointed out that is was the same for the OBE and SSE in ASME Code Case N-411). The staff also found that the overall contribution of earthquake cycles to fatigue is small. To design for 20 full SSE cycles (or the equivalent) would not significantly penalize the design and would provide a bounding design for the expected number of earthquakes of a lesser magnitude than the SSE and their aftershocks for a 60-year plant life.

In its letter of September 16, 1992, ACRS stated that it believes the staff took an appropriate approach in its interim position (which has been incorporated into the final staff position above).

In its final position, the staff supplemented its preliminary positions from the draft Commission papers and provided a more complete package identifying the actions necessary for the design of SSC when the OBE is eliminated. As discussed above, the staff clarified that guidelines should be maintained to ensure the functionality of components, equipment, and their supports. In addition, the staff clarified how certain design requirements are to be considered for buildings and structures that are currently designed for the OBE earthquake, but not the SSE. Also, the staff addressed how pre-earthquake planning and post-earthquake operator actions are to be considered. The staff's proposed guidelines and their bases are discussed in the final positions above.

The staff has evaluated the effect on safety of eliminating the OBE from the design load combinations for selected SSC and has developed proposed criteria for an analysis using only the SSE. The staff, therefore, requests that the Commission approve the proposed 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 criteria different than those described above for an SSE-only analysis.

N. Inservice Testing of Pumps and Valves

In SECY-90-016, the staff recommended that the Commission approve the position that 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 that cannot be detected through the use of advanced, non-intrusive techniques.

The staff concluded that these requirements are necessary to provide an adequate assurance of operability.

In its SRM of June 26, 1990, the Commission approved the staff's position as supplemented in the staff's response to ACRS comments, dated April 27, 1990. In that response, the staff agreed with the ACRS recommendations to emphasize the requirements of Generic Letter (GL) 89-10 with regard to evolutionary plants, to resolve check valve testing and surveillance issues, and to indicate how these requirements are to be applied to evolutionary plants. The staff also agreed that the requirements should permit consideration of proposed alternative ways of meeting inservice and surveillance requirements. The Commission further noted that due consideration should be given to the practicality of designing testing capability, particularly for large pumps and valves.

After reviewing the proposed staff requirement in SECY-90-016 which stated that designs should incorporate provisions to test motor-operated valves under design-basis differential pressure, the staff has clarified its position. The staff recommends that full flow testing be conducted at maximum design flow with analysis to extrapolate to design pressure if it is not practicable to conduct the inservice pump testing at design flow and pressure. The staff also recommends that for valves, a qualification test (under design-basis differential pressure) be conducted prior to installation and inservice valve tests be conducted under the maximum practicable differential pressure and flow when it is not practicable to achieve design-basis differential pressure during an inservice test.

In its letter of May 5, 1992, EPRI stated that the ALWR program agrees with the above staff positions for the passive and evolutionary plants. In its letter of August 17, 1992, ACRS stated that they support the staff's recommendation that the above design, testing, and inspection provisions should be imposed on all safety-related pumps and valves for passive ALWRs.

EPRI and the evolutionary ALWR designers have indicated that their submittals are consistent with these criteria. The staff is in the process of evaluating their submittals to ensure acceptable implementation of the Commission's guidance on this issue. The passive ALWR vendors have indicated that their designs will comply with the applicable EPRI requirements document.

Therefore, the staff recommends that the Commission approve the position that these requirements should also be imposed on passive ALWRs. The staff concludes that additional inservice testing requirements may be necessary for certain pumps and valves in passive plant designs. This necessity arises because the passive safety systems rely heavily on the proper operation of this equipment (such as check valves or depressurization valves) to mitigate the effects of accidents and to shut down the reactor. The staff will discuss its proposed resolution of this issue in a separate Commission paper addressing the regulatory treatment of nonsafety systems in passive plant designs.

II. Other Evolutionary and Passive Design Issues

A. Industry Codes and Standards

In SECY-91-273, "Review of Vendors' Test Program to Support the Design Certification of Passive Light-Water Reactors," dated August 27, 1991, the staff raised the concern that a number of design codes and industry standards dealing with new plant construction have recently been developed or modified, and that the NRC has not yet determined their acceptability.

The staff recommends that the Commission approve the position, consistent with past practice, that it will review both evolutionary and passive plant design applications using the newest codes and standards that have been endorsed by the NRC. Unapproved revisions to codes and standards will be reviewed on a case-by-case basis. In its letter of May 13, 1992, ACRS agreed with the staff's position. Similarly, in its letter of May 5, 1992, EPRI stated that the staff's position is consistent with the EPRI requirements documents.

B. Electrical Distribution

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," dated March 25, 1991, the staff recommended that the Commission approve its position that an evolutionary plant design should include the following elements:

- an alternative 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,

- 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 SRM of August 15, 1991, the Commission approved the staff's positions.

In its letter of May 5, 1992, EPRI indicated that this issue is not applicable to passive designs. However, the staff has not yet determined the applicability of this issue to the passive designs. This issue will be addressed in a separate Commission paper which will discuss the regulatory treatment of active nonsafety systems in passive plant designs.

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 PRA. Current regulations do not require that a seismic PRA be performed to determine if a site is acceptable, and the staff does not intend to require such an assessment.

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). For this comparison, the staff used 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 also 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. Based on 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.

Similarly, the LLNL hazard curves used in the staff's reviews of seismic hazard 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 results using both 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 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 review of historical seismicity and the LLNL hazard estimates, the staff concludes 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.

To judge the seismic capability of the GE and ABB-CE designs for sites in the continental U.S., the staff used a deterministic process. On that basis, the staff concludes that, with few exceptions, most areas of the U.S. 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 are within the bounding site parameters for the certified design.

In its letter of May 5, 1992, EPRI stated that they now specify a Seismic Margins Assessment (SMA) methodology, which plant designers can use to assess the capability of advanced plants to shut down following a seismic event greater than an SSE. In addition, the PRA methodology may be used in evaluating a balanced seismic capability of standard designs, but calculation of risk as part of an overall core damage frequency determination is not required. The staff has reviewed this SMA methodology as part of its review of the EPRI requirements documents and should resolve any issues associated with this methodology through resolution of open items in the safety evaluation reports.

The discussion on seismic hazard curves provided in this section is for information only. 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.

D. Leak-Before-Break

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, dated 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. However, it must first be 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 flaws could grow to unstable sizes. The concept underlying such analyses is referred to as "leak-before-break" (LBB).

To date, the NRC staff has approved the LBB approach 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 use of LBB for 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 of at least 6 in. nominal diameter. The piping includes both austenitic and carbon steel material. However, all of the LBB-approved carbon steel piping has been clad with stainless steel material. To date, no BWRs have requested LBB approval.

EPRI and GE have proposed to adopt the LBB approach for ALWRs when certain details of the piping design, material properties, and stress conditions are known. As discussed in SECY-89-013, "Design Requirements Related to the Evolutionary Advanced Light Water Reactors (ALWRs)," dated January 19, 1989, the staff will evaluate the acceptability of the use of LBB 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 that both upper- and lower-bounding limits are determined. Such limits would ensure that adequate margins are available for leakage, loads, and flaw sizes.

A leakage margin of 10 is required to ensure that leakage from the postulated flaw size is detected when the pipe is subjected to normal operational loads. A load margin of 1.4 is required to ensure that leakage-size flaws are stable at normal plus accident loads (such as SSE 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 NRC 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 SSE 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 also required. An applicant may propose any fracture mechanics evaluation method for NRC staff review. However, the applicant will have to demonstrate the accuracy of the method by comparing it with other acceptable methods or with experimental data.

Using this 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 becomes available during the COL phase. Before fuel-loading, the preliminary LBB analyses should be completely verified and based on actual material properties and final, as-built piping analysis as part of the inspections, 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 NRC has established certain limitations for excluding from the LBB approach piping that is likely to be susceptible to failure from various degradation mechanisms during service. A significant portion of the LBB review involves evaluating 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 NRC staff reviews the operating history and measures to prevent or mitigate these mechanisms.

The LBB approach cannot be applied to piping that can fail in service from such effects as water hammer, creep, 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 of a geometry that 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 and corrosion have been eliminated through the use of high chromium steels with proven resistance to erosion and corrosion or through the use of carbon steel piping that is clad on the fluid-contacting surface with materials resistant to erosion and corrosion.

A detailed discussion of the limitations and acceptance criteria used for LBB 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 the containment, ECCS, 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 generally be divided into local dynamic effects and global effects. Local dynamic effects are uniquely associated with a particular pipe rupture. These specific effects are not caused by a failure of 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 of such sources as pump seals, leaking valve packings, flanged connections, bellows, manways, rupture disks, and 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 (such as safety-relief valve discharges, pool-swell/fallback, condensation oscillation, and chugging loads).

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). Nonetheless, the possibility of such dynamic effects being caused by a reactor internal pump ejection, failures of flanged connections, and blowdowns from ruptured 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 or feedwater line break to envelop the possible global dynamic effects described above. The

designer would be required to submit this approach for NRC staff review and approval before 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, and design of subcompartment enclosures.

*Recommendations*

The revised GDC 4 of Appendix A to 10 CFR Part 50 permits elimination of the dynamic effects of postulated high-energy pipe ruptures from the design basis of ALWRs using advanced fracture mechanics analyses. 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. This approval should be limited to instances in which appropriate bounding limits are established using preliminary analysis results during the design certification phase and verified during the COL 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.

In its letter dated May 13, 1992, ACRS agreed with the staff's recommendation to extend the application of the LBB approach to both evolutionary and passive ALWR plants. Similarly, in its letter dated May 5, 1992, EPRI agreed to extend the LBB approach to both evolutionary and passive plants and has since revised its URD to be consistent with the final staff's position stated above. The staff has not revised this position on LBB since the draft Commission papers.

E. Classification of Main Steamlines in Boiling Water Reactors (BWR)

The main steamlines in BWR plants contain dual quick-closing main steam isolation valves (MSIVs). These valves 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. This calculation is based on a conservative assumption that the leakage 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, the staff developed guidance in RG 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants." This RG recommends the installation of a supplemental leakage control system (LCS) to ensure that the isolation function of the MSIVs complies with the specified limits. However, operating experience has shown that the LCS has required substantial maintenance and resulted in substantial worker radiation 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 allow higher leakage limits through the MSIVs, eliminate the safety-related leakage control system, and use an alternative MSIV leakage treatment method that takes credit for the large volume and surface area in the main steam piping and condenser hotwell to plate-out the fission products following core damage. In this way, the main steam piping, main steam drain and bypass lines, and condenser are used to mitigate the consequences of an accident.

Appendix A to 10 CFR Part 100 requires that SSCs necessary to ensure the capability to mitigate the consequences of accidents remain functional during and after a SSE. These components are classified as safety-related and seismic Category I. In addition, Appendix B to 10 CFR Part 50 establishes QA requirements for safety-related, seismic Category I SSCs.

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). RG 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 should conform to Appendix A to SRP Section 3.2.2 and RG 1.29, as should branch lines 2½ inches in diameter and greater up to the first closed valve. 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 if it has been dynamically analyzed to demonstrate structural integrity under SSE loading conditions. However, all pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 apply to this portion of the main steamline from the seismic interface restraint to the turbine stop valve. These requirements are needed to ensure that the quality of the piping material is commensurate with its importance to safety during both operational and

accident conditions. In accordance with Position C.1.e of RG 1.29, the turbine stop valve shall be designed to withstand the SSE and maintain its integrity.

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, but not including, the turbine stop valve are equivalent to the staff guidelines in Appendix A to SRP Section 3.2.2 and RG 1.29.

The staff and EPRI 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. In addition, this would ensure the integrity of the main steam drain lines and bypass piping from the first valve to the main condenser hotwell. The remaining issue is the classification of the main steam drain lines and bypass piping between the first normally-closed valve and the condenser hotwell as well as the hotwell itself. The staff proposes that neither the main steam drain and bypass lines from the first valve up to the condenser inlet, nor the piping between the turbine stop valve and the turbine inlet should be classified as safety-related or as seismic Category I. Rather, these lines should be analyzed using a dynamic seismic analysis to demonstrate structural integrity under SSE loading conditions. The turbine stop, control, and bypass valves and the main steam lines from the turbine control valves to the turbine shall meet all of the quality group and quality assurance guidelines specified in SRP Section 3.2.2, Appendix A.

Further, the staff proposes that seismic analyses be performed to ensure that the condenser anchorages and the piping inlet nozzle to the condenser are capable of maintaining their structural integrity during and after the SSE. The dose analysis considers that the condenser is open to the atmosphere. Thus, it is only necessary to ensure that gross structural failure of the condenser will not occur. Similarly, it is only necessary to ensure that failure of non-safety-related SSC resulting from a seismic event will not cause failure of the main steam piping, main steam drain and bypass lines, or condenser.

In its letter of May 13, 1992, the ACRS stated that it agrees with the staff's recommendation for the main steamline classification for both evolutionary and passive BWRs. In a letter dated May 5, 1992, EPRI stated that it agrees with the staff's recommendations and proposed several clarifications, which the staff has incorporated into the above discussion.

The staff concludes that the above-described approach to resolve the BWR main steamline issue for both evolutionary and passive ALWRs provides reasonable

assurance that the main steam piping from the outermost isolation valve up to the turbine stop valve, the main steam drain and bypass lines up to the condenser, and the main condenser will retain their pressure and structural integrity during and following a SSE. The staff recommends that the Commission approve the above-described approach to resolve the main steamline classification for both 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 RG 1.76 "Design Basis Tornado for Nuclear Power Plants." According to WASH-1300, the probability of occurrence of a tornado that exceeds the design basis tornado (DBT) should be on the order of $10^{-7}$ per year for each nuclear power plant. The regulatory guide delineates maximum wind speeds of 386 to 597 kilometers per hour (km/hr) (240 to 360 miles per hour (mph)) depending on the regions.

As a result of EPRI's earlier efforts on this EPRI-proposed plant optimization subject, the NRC reevaluated the regulatory positions in RG 1.76 using the considerable quantity of tornado data which has become available since the regulatory guide was developed. The reevaluation is 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), which contains data for the 30 year period from 1954 through 1983. This tape contains data for the approximately 30,000 tornados that occurred during the period.

The staff determined that the tornado strike probabilities range from near $10^{-7}$ per year for much of the western United States to about $10^{-3}$ 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 $10^{-7}$ per year were estimated to be about 322 km/hr (200 mph) for states west of the Rocky Mountains, and 482 km/hr (300 mph) for states east of the Rocky Mountains.

In its letter of December 6, 1991, EPRI proposed that the design-basis tornado to be used in the design of evolutionary ALWRs be based on a maximum tornado wind speed of 482 km/hr (300 mph) and a tornado strike probability derived from a recurrence interval of $10^{-7}$ per year. During a meeting with the staff on January 30, 1992, 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 to establish structural requirements (such as minimum concrete wall thicknesses) to protect nuclear plant safety-related SSC against effects not explicitly addressed in review guidance (such as RG or the SRP). Specifically, the staff has routinely reviewed and evaluated aviation crashes (involving general aviation

light aircraft), nearby explosions, and explosion debris or missiles, taking into account the tornado protection requirements. The staff's acceptance of EPRI's proposal will 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 322 km/hr (200 mph) for states west of the Rocky Mountains, and 482 km/hr (300 mph) for states east of the Rocky Mountains. Therefore, the staff recommends that the Commission approve the position that a maximum tornado wind speed of 482 km/hr (300 mph) be used in the design-basis tornado employed in the design of evolutionary and passive ALWRs. The COL applicant will have to demonstrate that a design capable of withstanding a 482 km/hr (300 mph) tornado will also be sufficient to withstand other site-specific hazards.

In its letter of May 13, 1992, ACRS agreed with the staff that the best available data should be used to establish the tornado design basis. However, ACRS noted the need to account for other potential loads that may previously have been subsumed within the tornado design-basis. The staff's position in the final policy paper has not changed, and the COL applicant will be required to demonstrate that the tornado design envelopes site-specific external impact hazards.

In its letter of May 5, 1992, EPRI agreed with the staff's position to use a 482 km/hr (300 mph) maximum tornado wind speed and to consider other site-specific hazards in the COL or early site permit.

The staff expects that the use of these 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, the COL applicant would have the option of performing a site specific analysis to verify that the design is still acceptable for that site.

G. Containment Bypass

The phenomenon of containment bypass is associated with either 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 released from the reactor coolant system will be condensed (thereby limiting the pressure increase in the containment) and scrubbed of radionuclides in a pool of water (the suppression pool). This is accomplished by directing the steam from the reactor coolant system to the suppression pool through a vent system. However, leakage paths could exist (between the drywell and the wetwell airspace) that could allow steam to bypass the suppression pool and pressurize (or over-pressurize) the containment.

Potential sources of steam bypass include vacuum relief valve leakage, cracks in the drywell structure, and penetrations through the drywell structure. Therefore, the staff concludes that vendors should make reasonable efforts to minimize the possibility of bypass leakage and should account, in their containment designs, for a certain amount of bypass leakage.

In addition, for a containment design that uses an external heat exchanger, the potential exists 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. Containment sprays in the drywell or wetwell would reduce the effect 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 ALWR designs. The GE ABWR and the ABB-CE System 80+ evolutionary designs have containment spray systems and, therefore, this issue is resolved for evolutionary designs.

In its letter of May 5, 1992, EPRI stated that they believed three separate topics were included in this issue. These topics were (1) interfacing system LOCA; (2) bypass of the suppression pool (BWR); and (3) failure of heat exchanger tubes in the passive containment cooling system (BWR). EPRI also indicated that their requirements documents adequately address these issues.

The staff will address the issue of containment bypass and whether passive designs should contain a containment spray system in a separate Commission paper which will discuss resolution of issues related to source term.

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 interval currently required by Appendix J to 10 CFR Part 50. This new maximum interval would apply to both evolutionary and passive plant designs. EPRI proposed this modification 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 (such as air lock testing and Type C leak testing methods) have also been raised, but have not been forwarded to the Commission as policy questions.

In parallel with the staff's review of this ALWR issue, the staff has developed proposed changes to Appendix J to 10 CFR Part 50 for all reactors. These changes were transmitted to the Commission in SECY-91-348, "Issuance of Final Revision to Appendix J to 10 CFR 50, and Related Final RG 1.XXX (MS 021-5)." This document proposes modification of the regulation that would allow the increased interval and addresses other issues raised by EPRI. SECY-91-348 also presents the staff's justification for these modifications.

This proposed rulemaking was withdrawn and is currently being reexamined under the Commission-approved program "Elimination of Requirements Marginal to Safety." The staff expects to complete this re-examination and advise the Commission on its final decision about the rulemaking course of action in December 1993.

In addition, the staff has extended (by as much as 1 year) the time interval for performing Type C leakage rate testing on currently operating plants on a case-by-case basis.

In its letter of May 13, 1992, ACRS identified no significant safety penalty caused by this change to the maximum interval between Type C leakage rate tests and agreed with the proposed staff position. In its letter of May 5, 1992, EPRI recommended that the Commission approve the staff's position.

Therefore, the staff recommends that the Commission approve the position that, until the rule change proceedings for Appendix J of 10 CFR Part 50 are completed, the maximum interval between Type C leakage rate tests for both evolutionary and passive plant designs should 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 (such as noble gases, iodines and cesiums, and non-volatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations.

RG 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 10 CFR 50.34(f)(2)(viii).

EPRI has proposed deviation from several aspects of the PASS design requirements, as discussed below. EPRI has identified this issue as a plant 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 Item II.F.1 of NUREG-0737 "Clarification of TMI Action Plan Requirements." The staff concludes that the safety-grade instrumentation provides adequate capability for monitoring post-accident

hydrogen, and this is acceptable justification for an ALWR vendor to use in requesting this deviation. Because this exemption has previously been granted on currently operating plants, the staff does not consider this request to be a policy matter.

In its letter dated May 13, 1992, ACRS agreed that elimination of the hydrogen analysis of containment atmosphere samples is appropriate, given that the safety grade hydrogen monitoring instrumentation will be installed.

*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 venting will remove gases accumulated in the reactor vessel (mainly hydrogen), and because prompt depressurization and cooling will minimize corrosion resulting from the presence of chloride and oxygen. 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)(viii) and Item II.B.3 of NUREG-0737 specify that the PASS should have the capability to analyze dissolved gases and chloride. This requirement was formulated before reactor vessels were required to have vents. With vented reactor vessels, the information on dissolved gas concentration became less important.

The staff concludes, however, that in PWRs even with vented reactor vessel some postulated accident sequences can occur in which the reactor coolant system is intact at reduced pressure, and heat is removed by subcooled decay heat removal (as in the TMI-2 accident). For these cases, it will not be possible to evaluate concentrations of the dissolved gases in reactor coolant by measuring their concentrations in the containment. For PWRs exposed to these conditions, information on the amounts of dissolved hydrogen 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. Therefore, the staff concludes that for PWRs the requirement for PASS sampling 24 hours after the accident would be adequate to help ensure long-term decay heat removal.

In BWRs whenever core uncovering is suspected, the reactor vessel is depressurized to within approximately the pressure within the wetwell and the drywell. As a result of this decrease in pressure, dissolved gases will partially pass out of the water phase into the gas phase and under these conditions the concentration of dissolved gases in the reactor coolant would have little meaning. During accidents in which only a small amount of cladding damage has occurred and the reactor vessel has not been depressurized, pressurized reactor water samples may be obtained from the process sampling system. Information on chloride and oxygen concentrations in the reactor coolant, although helpful in ensuring that proper steps are taken to minimize corrosion of reactor components, constitutes a secondary

consideration since these samples could be taken at a low pressure. Therefore, for BWRs, there will be no need for taking pressurized samples.

In its letter of May 5, 1992, EPRI stated that sampling of the reactor coolant for dissolved gases and chlorides is not needed because ALWR coolant depressurization systems are highly reliable. Further, EPRI indicated that significant requirements associated with prevention of severe accidents make the need for PASS exceedingly small, especially in view of the limited usefulness to the operators of the information obtained from these measurements. In addition, even if a need for this information develops, EPRI maintained that low levels of fuel failures predicted in ALWRs would permit operators to use routine sampling equipment.

In its letter to ACRS dated June 12, 1992, the staff provided the justification for retaining this sampling equipment for PWRs. The need for the dissolved gas sampling stems from the possibility of partially mitigated severe accidents which do not involve early reactor depressurization, as was demonstrated in the TMI-2 accident. In addition, there is a concern that these reactors may have a problem in maintaining reliable natural circulation and decay heat removal in the presence of non-condensible gases that would evolve during depressurization. This concern is especially important in passive PWRs where decay heat removal systems are highly dependent on natural circulation. Also, in these reactors, non-safety systems would need to be used to perform the final cooldown depressurization. These actions would require that the operator fully appreciate the consequences of depressurizing the plant and possibly introducing non-condensible gases that may interfere with the successful termination of the event.

Determination of chloride concentrations, although helpful in ensuring that plant personnel take appropriate actions to minimize the likelihood of accelerated primary system corrosion following the accident, is a secondary consideration since long term samples could likely be taken at a low pressure. It does not constitute, therefore, mandatory requirement of the PASS.

Therefore, the staff recommends that the Commission approve the position that post-accident sampling systems for evolutionary and passive ALWRs of the pressurized water reactor type be required to have the capability to analyze dissolved gases and chloride, in accordance with the requirements of 10 CFR 50.34(f)(2)(viii) and Item III.B.3 of NUREG-0737. The time for taking these samples can be extended to 24 hours following the accident. For evolutionary and passive ALWRs of the boiling water reactor type, there would be no need for the post-accident sampling system to analyze dissolved gases.

*Relaxation in the Time Requirement for Sampling Activity Measurements*

EPRI states in the Passive Requirements Document that if boron solution has been added to permit plant shut down, reactor water samples can be taken for boron analyses starting eight hours after the end of power operation. EPRI also states that the samples for activity measurements will not be required until the accident recovery phase.

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 following the accident. The purpose of this requirement is to ensure the capability to draw samples while the accident is still in progress, because analyses of the samples can provide insights for accident mitigation measures. EPRI has committed that the neutron flux monitoring instrumentation will comply with the Category I criteria of RG 1.97. Therefore, this instrumentation will have fully qualified, redundant channels that monitor over the power range of $10^{-6}$ percent to full power. Based on this commitment, the staff concurs with EPRI's assertion that sampling for boron concentration measurements will not be required for the first 8 hours after an accident.

By contrast, samples for activity measurements are required to evaluate 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 arise 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 time extension for sampling activity measurements to 24 hours after an accident is acceptable.

Therefore, the staff recommends that the Commission approve the deviation from the requirements of Item II.B.3 of NUREG-0737 with regard to requirements for sampling reactor coolant for boron concentration and activity measurements using the post-accident sampling system in evolutionary and passive ALWRs.

The modified requirement would require the capability to take boron concentration samples and activity measurements 8 hours and 24 hours, respectively, following the accident.

J. Level of Detail

In its SRM of February 15, 1991, concerning SECY-90-377, "Requirements for Design Certification Under 10 CFR Part 52," 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 this 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 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 staff has informed the Commission of the progress of its efforts to resolve the level of detail issue and has requested Commission guidance when appropriate. The following Commission papers have addressed issues related to level of detail:

- SECY-91-178, "Inspections, Test, Analyses, and Acceptance Criteria (ITAAC) for Design Certification and Combined Licenses;"
- SECY-92-053, "Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews;"
- SECY-92-196, "Development of Design Acceptance Criteria (DAC) for the Advanced Boiling Water Reactor (ABWR);"
- SECY-92-214, "Development of Inspections, Test, Analyses, and Acceptance Criteria (ITAAC) for Design Certification;" and,
- SECY-92-287, "Form and Content for a Design Certification Rule."

The staff concludes that the level of detail issue is applicable to all design certification applications, but expects to resolve this issue 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 is seeking no further guidance from the Commission on this issue.

K. Prototyping

In SECY-91-074, "Prototype Decisions for Advanced Reactor Designs," the staff discussed the process it will use to assess the need for a prototype or other demonstration facility for the advanced reactor designs. The staff stated that it will follow the procedure outlined in SECY-91-074 to identify 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, it cannot be resolved during the EPRI review. The staff has evaluated the submittals of evolutionary ALWRs to assess the need for prototype testing and has not identified any areas that may require such testing.

As discussed in SECY-91-273, "Review of the Vendor's Test Programs to Support the Design Certification of Passive Light-Water Reactors," the need for separate effects and scaled integral testing for passive designs is under consideration. The staff is currently reviewing vendor test programs for the Westinghouse AP600 and the GE SBWR and working closely with the vendors to resolve concerns. Since the issuance of SECY-91-273, the staff has forwarded to the Commission several Commission papers discussing integral system testing requirements and proposed NRC-sponsored confirmatory testing for the AP600 and SBWR designs.

The prototyping discussion 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.

L. ITAAC

ITAAC are required for certified designs in accordance with Subpart B of 10 CFR Part 52. Licensees that reference a certified design will implement the related ITAAC. The Nuclear Management and Resources Council (NUMARC) has designated GE as the industry lead for developing ITAAC on their ABWR application. The staff is working with GE and NUMARC to develop ITAAC for design certification and has kept the Commission apprised of the status of ITAAC. The following ITAAC-related papers have been sent to the Commission:

- SECY-91-178, "Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Design Certifications and Combined Licenses," June 12, 1991;
- SECY-91-210, "Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Requirements for Design Review and Issuance of a Final Design Approval," July 16, 1991;
- SECY-92-053, "Use of Design Acceptance Criteria (DAC) During 10 CFR Part 52 Design Certification Reviews," February 19, 1992;
- SECY-92-196, "Development of Design Acceptance Criteria (DAC) for the Advanced Boiling Water Reactor (ABWR)," May 28, 1992;
- SECY-92-214, "Development of Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Design Certifications," June 11, 1992;
- SECY-92-299, "Development of Design Acceptance Criteria (DAC) for the Advanced Boiling Water Reactor (ABWR) in the Areas of Instrumentation and Controls (I&C) and Control Room Design," August 27, 1992; and,
- SECY-92-327, "Reviews of Inspections, Test, Analyses, and Acceptance Criteria (ITAAC) for the GE Advanced Boiling Water Reactor (ABWR)," September 22, 1992.

These papers discuss a wide range of issues related to ITAAC including how non-traditional issues discussed in this paper such as severe accidents and insights from the PRA are included in ITAAC. 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.

M. Reliability Assurance Program

In SECY-89-013, the staff stated that a program called the reliability assurance program (RAP) would be required for design certification to ensure that the design reliability of safety significant SSC is maintained over the life of a plant. In November 1988, the ALWR vendors and EPRI were informed that the staff was considering this matter.

The ALWR RAP would apply to those plant SSCs that are risk-significant (or significant contributors to plant safety) as quantified by the design certification PRA. The purposes of the RAP are (1) to ensure that an ALWR is designed, constructed, and operated in a manner that is consistent with the design assumptions for these risk-significant SSCs, (2) to prevent the reliability of these risk-significant SSCs from degrading during plant operations, (3) to minimize the frequency of transients that challenge ALWR SSCs, and (4) to help ensure that these SSCs function reliably when challenged.

The staff views the RAP for ALWRs as a two-stage program. The first stage applies to the design phase of the plant life cycle, and is referred to as the design reliability assurance program (D-RAP). The second stage applies to the construction and operations phases of the plant life cycle and is referred to as the operational reliability assurance program (O-RAP). An applicant for design certification shall be required to establish the framework for the RAP (e.g., scope, purpose, objective, and essential elements of an effective RAP) and shall implement those portions of the D-RAP that apply to design certification. A COL applicant would augment the design certification D-RAP with site-specific design information and would implement the balance of the D-RAP, including input to the procurement process. The COL applicant would also establish and implement the O-RAP.

The O-RAP can be thought of as an umbrella program that would integrate aspects of existing programs (e.g., maintenance, surveillance testing, inservice inspection, inservice testing, and QA) to achieve its objective. The O-RAP would apply to the construction and operation phases of plant life. To ensure regulatory coherence between RAP and the maintenance rule (10 CFR 50.65), performance goals for risk-significant SSCs would be established under the O-RAP based on input from the D-RAP. As such, performance and condition monitoring requirements for maintaining the reliability of risk-significant SSCs would be established. In addition, O-RAP would provide a feedback mechanism for periodically re-evaluating risk significance based on actual equipment, train, or system performance. The majority of the O-RAP would be based on the requirements of 10 CFR 50.65.

The staff has completed its review of the RAP in the EPRI URD for the evolutionary ALWR, as documented in the final safety evaluation report (FSER) for Chapter 1, Section 6. The staff has also documented the results of its review of the GE ABWR D-RAP in the draft final safety evaluation report (DFSER) for Chapter 17.3. The staff has had the benefit of discussions with the ACRS regarding the form and content of the ALWR RAP. In its letter dated

October 15, 1992, concerning proposed guidance for implementation of the maintenance rule (10 CFR 50.65), the ACRS noted the similarity between the maintenance rule, the license renewal rule, and the RAP. The ACRS stated that consistent staff guidance is needed on the elements of an acceptable program that will satisfy these three sets of requirements. The staff has incorporated the ACRS comments in developing its position on the RAP.

The following section summarizes the position the staff has taken in performing its review of D-RAP submittals contained in the design certification applications and the EPRI URD for the evolutionary and passive ALWRs.

*Interim Staff Position on RAP*

For design certification, the staff position is that a high level commitment to a RAP applicable to design certification (D-RAP) should be required as a non-system generic Tier 1 requirement with no associated ITAAC. The details of the D-RAP, including the conceptual framework, program structure, and essential elements, should be provided in the SSAR. The SSAR for the D-RAP should also (1) identify and prioritize a list of risk-significant SSCs based on the design certification PRA and other sources, (2) ensure that the vendor's design organization determines that significant design assumptions, such as equipment reliability and unavailability, are realistic and achievable, (3) provide input to the procurement process for obtaining equipment that satisfies the design reliability assumptions, and (4) provide these design assumptions as input to the COL for consideration in the operational reliability assurance program (O-RAP). A COL applicant would augment the design certification D-RAP with site-specific design information and would implement the balance of the D-RAP, including input to the procurement process. The COL applicant would also establish and implement the O-RAP. The staff will review the COL applicant's D-RAP and O-RAP as part of the COL application.

The staff position on O-RAP will be provided in a future Commission paper. In developing its position on O-RAP, the staff will ensure that regulatory coherence exists between O-RAP and the requirements of the maintenance rule and the license renewal rule.

The staff's proposed resolution of the RAP for design certification will be provided to the Commission in a separate Commission paper which will discuss the regulatory treatment of active non-safety systems in passive plants. The discussion provided in this section is for information only and is provided to identify a complete list of issues applicable to evolutionary and 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.

N. Site-Specific Probabilistic Risk Assessments and Analysis of External Events

In its "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 PRA. The Commission also stated that applicants should 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. Further, the Commission stated that evolutionary plant vendors should use the PRA in considering a range and combination of alternatives that address unresolved and generic issues and to search for cost-effective means to reduce the risk associated with severe accidents. In the policy statement, the Commission directed the staff to 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 promulgated 10 CFR Part 52. Section 52.47 requires that an application for design certification contain a design-specific PRA.

In GL 88-20, "Individual Plant Examinations for Severe Accident Vulnerabilities - 10 CFR 50.54(f)," and its supplements, the staff stated that construction permit holders and power reactor licensees should consider the safety implications of both internal and external events. Such consideration should involve 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 52.47 should include an assessment of both internal and external events.

Lessons from past risk-based studies indicate that fire, internal floods, and seismic events can be important potential contributors to core damage. However, the estimates of core damage frequencies for fire and seismic events continue to include considerable uncertainty. Consequently, the staff concludes that fire and seismic events can best be evaluated using simplified probabilistic methods and margins methods similar to those developed for existing plants, supported by insights from internal event PRAs (including ALWR design-specific PRAs). The designer should use traditional probabilistic techniques to study internal floods.

Fire events can be evaluated using simplified methods such as EPRI's Fire Induced Vulnerability Evaluation (FIVE) methodology rather than full scale PRAs. Ascribing to these methods, the designer would focus on the capacity of the design to withstand the effect of fire, using qualitative and quantitative methods rather than a strictly quantitative PRA fire analysis.

The staff concludes that the plant designer can best determine the seismic capability of the plant through a combined approach that takes advantage of

the strengths of both PRA and margins methods. This approach (based on an internal events PRA, its existing event and fault trees, and its random failures and human errors) allows for a comprehensive and integrated treatment of the plant's response to an earthquake. This approach should yield meaningful measures of a proposed design's seismic capability.

The major difference between a seismic PRA and the proposed PRA-based margins approach is that the latter does not convolute fragility curves with hazard curves. Rather, the PRA-based margins approach measures the robustness of the plant to withstand earthquakes of a given g-level. This method eliminates the need to deal with uncertainty in the seismic hazard curve for the site and identifies potential design-specific seismic vulnerabilities. Understanding these vulnerabilities may be useful in developing the reliability assurance programs, identifying operator training requirements, and focusing on accident management capabilities.

The staff will require each plant designer to perform a PRA-based margins analysis to identify the vulnerabilities of their design to seismic events. The plant designer should construct plant logic models covering the various systems that could be used to prevent core damage. Typically, this would be accomplished by modifying design-specific 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 then determine all important accident sequences using the event trees and fault trees (based on fragility data for each component for each sequence). In addition, the designer would determine the value of the minimum high confidence, low probability of failure (HCLPF) for the plant by determining the HCLPF values for the important SSCs for each accident sequences. The HCLPF values calculated in this manner can be used to measure the plant's robustness and to provide an acceptable estimate of the earthquake ground motion which the plant is expected to be able to survive without core damage. In general, the value of the plant HCLPF should be at least twice the design ground motion zero period acceleration. If not, the designer should perform a more detailed evaluation to identify any vulnerability against which the plant requires strengthened protection. HCLPF calculations also indicate which components and systems limit the seismic capability of the plant.

In its letter of May 8, 1992, EPRI stated that they were unaware of suitable margins approaches to evaluate external events other than a seismic margins assessment. In its letter of August 21, 1992, EPRI provided additional comments regarding the staff's proposed requirements for analysis of certain internal and external events. EPRI concluded that their requirements for separation of redundant SSE by physical barriers designed to withstand full-height compartment flooding should make core damage events resulting from internal flooding very unlikely. In addition, EPRI concluded that the proposed requirements for analyses of internal fire and seismic events would be addressed satisfactorily by their requirements documents, except for the provision of a seismic margin value twice the magnitude of the SSE. EPRI recommended that the seismic margin value be changed to 1.5 times the magnitude of the SSE to be consistent with operating plants. In addition,

EPRI noted that the as-built verification phase of the seismic margins evaluations can be based upon applying the appropriate margin to the site-specific SSE.

The staff concludes that separated safety divisions do not necessarily preclude internal floods from being significant contributors to risk. The staff notes that evolutionary ALWR designs that include physical separation between safety divisions appear to respond better to internal fires and floods than do traditional designs. However, although this physical separation should reduce the expected frequency and severity of these events, it does not preclude them from being important contributors to risk. For example, an internal flooding PRA study for an evolutionary design has already identified several design improvements that are needed to achieve the desired level of internal flood protection. The staff believes that the systematic evaluation required during the development of an internal flooding PRA is necessary to ensure that potential flood vulnerabilities are identified.
The EPRI seismic margins methodology limits the search for success paths to two, and will not provide some important insights concerning human error and random failure. Therefore, this methodology does not appear adequate for evaluating evolutionary and passive ALWR designs. In addition, the EPRI margins method is limited in its useful application for passive plant margins analyses as is the NRC margins method used in the IPEEE, since these margins approaches are based on insights from PRAs of operating plants (PWRs and BWRs). The staff concludes that it would be inappropriate to rely exclusively on either of these methods to perform seismic margins analyses for passive designs.

In addition, the staff concludes that a well-designed plant should have a plant HCLPF at least twice the magnitude of its SSE. Therefore, if the PRA-based margins analysis only looks at accelerations up to 1.5 times the magnitude of the SSE, the analysis could effectively screen out potential design-specific seismic vulnerabilities. The staff would anticipate that analysis of a fully developed seismic PRA will identify seismic events as significant contributors to risk (perhaps the largest, given the significant estimated reduction in internal event core-damage frequency in evolutionary ALWRs). Therefore, it is important to fully understand potentially significant seismic vulnerabilities and other seismic insights. This information would be captured by a PRA-based seismic margins analysis that considers sequence-level HCLPFs and fragilities for all sequences leading to core damage or containment failures up to approximately twice the magnitude of the SSE.

Details of the specific site characteristics will likely not be available until the COL review stage. The staff intends to require a COL applicant to perform a site-specific PRA that addresses all applicable site-specific hazards (such as river flooding, storm surge, tsunami, volcanism, or hurricanes). The staff will review the site-specific PRA to ensure that no vulnerabilities are introduced by siting the standardized plant at a location where external hazards could pose an unacceptable or unanticipated risk.

In its letter of May 13, 1992, and in discussions with the staff, ACRS noted that the staff, an ALWR vendor, and a COL applicant may experience significant obstacles if design vulnerabilities from site-specific external events are discovered at the COL review stage. ACRS requested more information on how the staff proposes to deal with unacceptable findings, resulting from the site-specific PRA, identified during the COL application review process.

In SECY-92-287, "Form and Content for a Design Certification Rule," the staff proposed the appropriate processes to modify Tier 1 and Tier 2 design certification information. For example, if a site-specific PRA identifies a serious generic design flaw that meets the "adequate protection" threshold, the NRC can initiate rulemaking to amend the design certification rule. If the site-specific PRA identifies a site-specific design weakness, the COL applicant will have the option to request an exemption to the design certification rule to correct the deficiency.

However, the staff will require that ALWR vendors perform bounding analyses of site-specific external events likely to be a challenge to a plant. When a site is chosen, its particular siting characteristics can then be compared to those of the bounding analyses in order to minimize the potential for the site-specific PRA to identify significant site-specific weaknesses for the standard design. Before certifying the design, the staff will evaluate fires, internal floods, and other external events that are not site dependent and will evaluate submitted bounding analyses for site-specific external events.

Therefore, the staff recommends that the Commission approve the position that the analyses submitted in accordance with 10 CFR 52.47 should include an assessment of internal and external events. PRA insights will be used to support a margins-type assessment of seismic events. A PRA-based seismic margins analysis will consider sequence-level HCLPFs and fragilities for all sequences leading to core damage or containment failures up to approximately twice the magnitude of the SSE. Simplified probabilistic methods, such as EPRI's FIVE methodology, will be used to evaluate fires. Traditional probabilistic techniques should be used to evaluate internal floods.

Secondly, the staff recommends that the Commission approve the position that ALWR vendors should perform bounding analyses of site-specific external events likely to be a challenge to the plant (such as river flooding, storm surge, tsunami, volcanism, high winds, and hurricanes). When a site is chosen, its characteristics should be compared to those assumed in the bounding analyses to ensure that the site is enveloped. If the site is enveloped, the COL applicant need not perform further PRA evaluations for these external events. The COL applicant should perform site-specific PRA evaluations to address any site-specific hazards for which a bounding analysis was not performed or which are not enveloped by the bounding analyses to ensure that no vulnerabilities due to siting exist.

The COL applicant should submit the results of the comparison of the bounding analyses to the site characteristics and any site-specific PRA information to the NRC at the COL review stage. For the GE ABWR and the ABB-CE System 80+

designs, the staff will work with the vendors to ensure a fundamental understanding of the potential vulnerabilities of their designs to external events. In order to maintain the design certification review progress, the staff will encourage, but not require, the vendors to complete these bounding analyses before issuance of a design certification rule. If the vendors choose not to perform the bounding analyses, they run a risk that site-specific hazards may limit acceptable sites for the design. For passive ALWR vendors who have submitted an application for design approval to the NRC, the staff will require that these bounding analyses be submitted and reviewed by the staff before issuance of final design approval. For subsequent design approval of evolutionary and passive ALWRs, the staff will require that the applicant submit bounding analyses for external events with the application.

At the COL stage, the staff will review the site-specific characteristics to ensure that events enveloped by bounding analyses at the design stage or evaluated by a site-specific PRA have been properly addressed. The staff plans to conduct walk-down inspections to confirm that design commitments have been met.

O. Severe Accident Mitigation Design Alternatives

As discussed in SECY-91-229, "Severe Accident Mitigation Design Alternatives for Certified Standard Designs," dated July 31, 1991, 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.

In Limerick Ecology Action v. NRC, 869 F.2d 719 (3rd Cir. 1989), the U.S. Court of Appeals effectively required the NRC to include consideration of severe accident mitigation design alternatives (SAMDAs) in the environmental impact review performed as part of the operating license application for the Limerick Generation Station. A NEPA evaluation, in the form of an environmental impact statement that includes consideration of SAMDAs, is an essential element of an application for a COL under Subpart C of 10 CFR Part 52, for those applications that reference a design certified under Subpart B. In SECY-91-229, the staff presented several options concerning the treatment of SAMDA issues as they related to the certification of standard plant designs. The staff recommended that the Commission approve the following recommendations:

- address SAMDAs for certified designs in a single rulemaking;
- approve the staff's approach for considering the costs and benefits of the review of SAMDAs for standard plant design certification; and,

- approve the staff's proposal to advise applicants for a final design approval and design certification that they must assess SAMDAs and provide the rationale supporting their decisions.

In its SRM of October 25, 1991, the Commission approved the staff's recommendations and requested that they be kept informed on the staff's progress in evaluating the SAMDAs for final design approval and design certification applications.

Consistent with the third recommendation of SECY-91-229, the staff requested that ALWR vendors assess SAMDAs for their designs. This assessment and subsequent staff review is in addition to the safety consideration of severe accident issues discussed in this paper. The staff is currently addressing responses to this request 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.

P. Generic Rulemaking Related to Design Certification

In SECY-91-262, "Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light-Water Reactor (LWR) Designs," dated August 16, 1991, the staff provided the Commission with recommendations regarding generic rulemaking related to design certification. The staff recommended that the Commission:

- approve the staff's proposal to proceed with design-specific rulemaking through individual design certifications to resolve selected technical and severe accident issues for the ABWR and System 80+ designs; and,

- 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 and 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 passive plant designs. Certain generic rulemaking activities related to the evaluation 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. In addition, the staff plans to 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.

Q. Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems

Instrumentation and control (I&C) systems help 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. Specifically, digital I&C systems share more data transmission functions and shares more process equipment than their analog counterparts.

Redundant trains of digital I&C systems may share data bases (software) and process equipment (hardware). Therefore, a hardware design error, software design error, or software programming error may result in a common-mode or common-cause 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 NRC staff developed these concerns in SECY-91-292, "Digital Computer Systems for Advanced Light Water Reactors." Some of the major points in that paper are summarized as follows:

- Common mode failures could defeat the redundancy achieved by the hardware architectural structure, and 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 and common-cause failures are quality and diversity. Maintaining high quality will increase the reliability of both individual components and complete systems. Diversity in assigned functions (for both equipment and human activities) equipment, 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 RG 1.152, "Criteria for Programmable Digital Computer Software Systems in Safety-Related Systems of Nuclear Power Plants." However, as

discussed in SECY-91-292, there are currently no regulatory requirements that 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 standards for certifying the design of digital I&C systems for application in nuclear power plants. In Enclosure 2 of SECY-91-292, the staff discussed regulatory requirements that it is considering to help ensure defense against common-mode failures, including the following areas:

- assessment of diversity;
- engineering activities;
- design implementation; and,
- safety classification of I&C systems.

The staff has made significant progress in establishing regulatory guidance that could be used to assess diversity. With the support of LLNL the staff has performed a study of the GE 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 ABWR safety analysis report. The staff is using the results of this assessment to help determine the additional diversity necessary to defend against postulated common-mode software and hardware failures in the ABWR.

EPRI discussed requirements for engineering activities and design implementation for digital I&C systems in Chapter 10 of the EPRI ALWR for both evolutionary and passive plants. In SECY-91-292, the staff discussed the role of the EPRI URD as providing a frame of reference for the development of acceptance criteria for the design to adequately satisfy the requirements in the URD. The criteria needed to satisfy the requirements for engineering activities and design implementation would be developed by the staff using applicable national and international standards where available, or expert opinions where adequate standards have not been developed, during the review of specific ALWR designs.

As discussed in SECY-91-292, the staff is continuing to develop safety classification criteria for I&C systems in ALWR designs. The international technical community, through the International Electrotechnical Commission (IEC), recently issued an IEC standard, "The Classification of Instrumentation and Control Systems Important to Safety for Nuclear Power Plants." EPRI proposed certain classification positions in its "ALWR Position Paper for Passive System Classification and Requirements," submitted by a letter dated March 19, 1992. The staff is considering both of these documents before reaching a final position on safety classification criteria for I&C systems in

passive ALWR designs. The safety classification of digital I&C systems relates to diversity through the defense-in-depth assessment by crediting systems that have previously been classified as non-safety systems.

Recently, increased attention has been given to detailed assessments of the integrity of software applied to safety-critical functions. These assessments have covered a broad range of applications, including computer-based medical treatment facilities, computer-based fly-by-wire aircraft control systems, and nuclear power plant protection systems. The staff found a consensus among computer science and software engineering experts that such safety-critical applications should be backed-up by some system not based on software. The experts based this opinion on the facts that the quantitative estimate for the reliability of I&C systems based on high-integrity software cannot yet easily be determined. The type of this backup and the functions it should perform depend on the specific equipment and design features of the I&C system.

The EPRI ALWR requirements document places special emphasis on common-mode failures to ensure they are addressed in man-machine interface system (M-MIS) designs. EPRI stated that the ALWR Program has recognized from the onset that there is currently no accepted standard to accurately quantify software reliability. To offset this concern, the ALWR Program has emphasized the need for software quality and for a defense-in-depth approach to ensure the integrity of I&C functions including requirements for a backup hardwired manual actuation capability for system-level actuation of safety functions.

As previously discussed, the staff has established potential regulatory guidance to ensure adequate diversity for digital I&C system applications. As a result of its review, the staff proposed an approach, in the draft Commission paper dated June 25, 1992, for assessing the defenses against common-mode failures in a design. The proposed approach also specified requirements for a backup system which is not based on software and which is used for system-level actuation of critical safety functions and displays of safety parameters.

After carefully reviewing ACRS, industry, and vendor comments, the staff has developed a final position. The staff has concluded that analyses that demonstrate adequate, rather than equivalent, defense against the postulated common-mode failures would be allowed in the diversity assessment required of the applicant. The critical safety functions that require backup manual controls and displays would be specified. The staff would consider allowing more flexibility in implementing the requirements for an independent set of displays and controls. The necessary degree of flexibility depends on the specific equipment and design features of the I&C system and will be evaluated for each design. The intent is to permit the use of diverse digital equipment that is not affected by the identified common-mode failures and to reduce complexity in the design. The staff will not require only analog equipment and will consider allowing simple digital equipment.

As a result of these changes, the staff revised the initial position proposed in the draft Commission paper. The staff recommends that the Commission approve the following revised staff position:

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 adequately been addressed. The staff considers software design errors to be credible common-mode failures that must specifically be included in the evaluation. An acceptable method of performing analyses is described in NUREG-0493, "A Defense-In-Depth and Diversity Assessment of the RESAR-414 Integrated Protection System," March 1979. Other methods proposed by an applicant will be reviewed individually.
2. In performing the assessment, 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. For events postulated in the plant SAR, an acceptable plant response should not result in a non-coolable geometry of the core, violation of the integrity of the primary coolant pressure boundary, or violation of the integrity of the containment.

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 function. The diverse or different function may be performed by a non-safety system if the system is of sufficient quality to perform the necessary function under the associated event conditions. Diverse digital or non-digital systems are considered acceptable means. Manual actions from the control room are acceptable if adequate time and information are available to the operators. The amount and types of diversity may vary among designs and will be evaluated individually.

4. A set of safety-grade displays and controls located in the main control room shall be provided for manual, system-level actuation of critical safety functions and monitoring of parameters that support the safety functions. The displays and controls shall be independent and diverse from the safety computer system identified in items 1 and 3 above. The specific set of equipment shall be evaluated individually, but shall be sufficient to monitor the plant states and actuate systems required by the control room operators to place the nuclear plant in a hot-shutdown condition. In addition, the specific equipment should be intended to control the following critical safety functions: reactivity control, core heat removal, reactor coolant inventory, containment isolation, and containment integrity.

   The displays and controls shall be hardwired in the safety computer system architecture to the lowest practical level. To achieve

system-level actuation at the lowest level in the safety computer system architecture, the controls may be hardwired either to analog components or to simple, dedicated, and diverse software-based digital equipment that performs the system-level actuation logic. The safety parameter displays may include digital components exclusively dedicated to displays. This requirement would provide for an independent and diverse control logic for manual, system-level actuation of the safety function that would be connected downstream of the lowest-level safety software-based component without affecting the hardware (interconnecting cables and interfaces) between the lowest-level electronic cabinets and the plant's electromechanical equipment.

Human engineering principles and criteria shall be applied to the selection and design of the particular displays and controls. The design of the displays and controls shall ensure that the human system interface shall be adequate to support the human performance requirements.

Hardwired, system-level controls and displays provide the plant operators with unambiguous information and control capabilities. These 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 software failure of the digital safety 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.

R. Steam Generator Tube Ruptures

The staff has identified two distinct issues related to steam generator tube ruptures (SGTRs). These issues, involving multiple ruptures specific to passive PWRs and containment bypass potential resulting from SGTRs, are discussed below.

Multiple Steam Generator Tube Ruptures for Passive PWRs

A design-basis accident involving SGTR in the current generation of pressurized water reactors is a rupture of one steam generator (SG) tube, with a rate of discharge of primary coolant through the SG tube break greater than the normal charging capacity of the reactor coolant inventory control system. SRP Section 15.6.3 requires the applicant to conduct an analysis for a single SGTR, but there is currently no requirement to perform an analysis for multiple SGTRs. The staff is considering whether multiple SGTRs should be included in the plant design basis for advanced PWRs.

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," dated 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 $1.5 \times 10^{-2}$ per reactor year (RY). The staff based this estimate on the four events that occurred in approximately 300 "mature" reactor-years of Westinghouse plant operation in the U.S. ("mature" RYs are accumulated after the first 2 years of plant operation). ABB-CE and Babcock and Wilcox plants, which at that time had accumulated 77 and 66 mature RYs, respectively, 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 $1.6 \times 10^{-3}$ 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 RYs accumulated among all U.S. PWRs at that time.

Since the staff issued NUREG-0844, the total number of mature RYs 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. Experience with Westinghouse plants (6 SGTRs in 535 RYs) indicates that the frequency of a single SGTR is approximately $1.1 \times 10^{-2}$ per RY. ABB-CE and Babcock and Wilcox plants appear to have lower SGTR frequencies. With a failure rate of $1.1 \times 10^{-2}$ per RY, about 3 SGTRs should have statistically resulted in about 300 RYs of operating experience, and none have occurred. The NRC has not received any report of an multiple SGTR in any U.S. or foreign plant. For consistency with the NUREG-0844 estimate of the probability of multiple SGTR events, the staff has derived a new estimate based on a 50 percent confidence level for an event that has not occurred in approximately 827 RYs of U.S. PWR operation to date. This estimated frequency for an multiple SGTR event is approximately $8.4 \times 10^{-4}$/RY.

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 tubes. The probability estimates given above are for SGTRs that occur 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 four 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 an 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 had been severely damaged.

The staff is reviewing the issue of whether to consider a single SGTR or an 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 an multiple SGTR event could pose substantial challenges to the plant's passive safety systems.

In dealing with an SGTR in a conventional plant, operators isolate the faulted SG 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, leakage rate would likely increase with the number of tubes ruptured, and the operators would be required to 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 also 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 LOCA and this ADS permits injection of a large amount of low-pressure makeup water from the incontainment 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 the CMT level decreases.

The pressure of the primary system should be reduced to about that of the secondary system to inhibit primary-to-secondary leakage. However, primary system depressurization below that of the secondary system appears undesirable during an SGTR. 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 SGs 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

representatives indicate 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 SG, and terminate the event.

However, if an multiple SGTR occurs, with a substantially greater leakage rate of primary coolant, the AP600 might not be able to accommodate the accident without actuating 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 then result in secondary-to-primary leakage of unborated water. This water could flash to steam as it enters the RCS if the SG water is above the saturation temperature at the primary system pressure. Since the passive safety systems of the AP600 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 disrupt or degrade ECC injection. Therefore, contrary to the response of current plants, the plant may respond to an multiple SGTR event in a manner considerably different than that for a single SGTR. The consequences may also differ significantly.

The designer could provide a number of methods to minimize 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- and small-break LOCAs, have estimated frequencies of occurrence on the order of $10^{-3}$ to $10^{-5}$ per RY. These events generally provide the most rigorous test of the plant's safety systems. The staff recognizes, however, that the multiple SGTR frequency could be in the range of $10^{-3}$ to $10^{-4}$ per RY and that passive plant response for a multiple SGTR could significantly differ from that for a single tube rupture. This recognition 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 in the design basis for the AP600. As a minimum, the plant designer or applicant should analyze the single and multiple SGTR events 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 analyze multiple ruptures of two to five 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 URD require that future pressurized water reactors have substantially improved capability to handle SGTRs. EPRI also stated that these requirements address material, design, and operation improvements to prevent SGTRs, as well as design features to improve the performance and response of the plant after an SGTR. EPRI has also concluded that 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.

In its letter of September 16, the ACRS stated that it agreed with the staff's recommendation that Westinghouse should analyze the AP600 response to ruptures of up to five SG tubes and requested that the staff provide a "better technical basis" for estimating the frequency of multiple SGTRs.

In its letter to ACRS of October 22, 1992, the staff noted that the calculation for estimating the frequency of multiple SGTRs was not meant to represent a rigorous statistical analysis of multiple SGTR frequency. Rather, this calculation was intended to show that the approximate frequency of multiple SGTRs of a few tubes is on the same order of magnitude as that of other limiting faults. The frequency calculation, coupled with the unique response of the passive safety systems to accidents involving loss of primary inventory, has led the staff to conclude that evaluation of multiple SGTRs up to about five tubes is warranted. The staff will use the vendor's analyses and its own confirmatory analyses of these events to examine the response of the AP600 to an multiple SGTR. The staff will subsequently recommend the extent to which multiple SGTRs should be included within the design basis of the AP600.

In letters of August 21, 1992, and September 17, 1992, EPRI and Westinghouse representatives, respectively, provided comments concerning the staff's position on multiple SGTRs. Westinghouse and EPRI believe that the design basis for the AP600 should remain a single tube rupture, and that multiple tube ruptures should be analyzed on a safety margin basis, using best-estimate techniques. Westinghouse and EPRI also indicated that they expect analyses of multiple SGTRs involving up to five tubes to show that the AP600 can respond to these events without actuating the automatic depressurization system.

The staff does not share the views of Westinghouse and EPRI representatives. At this time, there is no experimental data on the performance of the AP600's passive safety systems that can be used to validate any models used for multiple SGTR analyses. The staff therefore believes that there is substantial uncertainty in any such analyses. If the ADS were to actuate in the course of an multiple SGTR, the subsequent interactions could have a serious impact on the ability of the passive systems to successfully terminate the accident. The staff expects that experimental data from planned vendor testing programs will provide an adequate basis for evaluating the response of the AP600 to an multiple SGTR.

The staff therefore maintains its stated position requiring analyses of multiple SGTRs up to five tubes for the AP600 and providing the results of

these analyses in the application for design certification. The staff will determine the appropriate number of SGTRs that should be included in the AP600 design basis after evaluation of the submitted analyses.

The staff recommends that the Commission approve the position to require that analysis of multiple SGTRs involving two to five 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.

Containment Bypass Potential Resulting From SGTRs

The staff has identified an additional containment performance issue that has not adequately been 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 its discussion on containment performance (Item I.J) in this Commission paper, the staff emphasized the importance of maintaining containment integrity following a postulated severe accident. In its SRM on SECY-90-016, the Commission endorsed the staff's goal of reducing the probability for conditional containment failure through the 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 attainment of the severe accident goals discussed in SECY-90-016.

The staff concludes that containment bypass resulting from SGTRs can be a significant challenge to containment integrity. Therefore, the staff concludes that the plant designer should consider design features that would reduce or eliminate containment bypass leakage in such a scenario. The following features could mitigate the releases associated with a tube rupture:

- 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; or,
- increased pressure capacity on the steam generator shell side with a corresponding increase in the safety valve setpoints.

In its letter dated September 16, 1992, the ACRS indicated that it agrees with the staff's recommendations to require that the applicant for design certification of a passive or evolutionary PWR assess design features necessary to mitigate the amount of containment bypass leakage that could result from multiple SGTRs.

In a letter dated August 21, 1992, EPRI indicated that the requirements document requires that future PWRs have substantially improved capability relative to SGTRs, to minimize the potential for containment bypass and avoid repetition of past incidents, where SGTR resulted in continued lifting of steam safety valves. As such, the plant design shall be such that the complete and sudden rupture of one steam generator tube will not result in actuation of steam side safety valves. EPRI indicated that this policy issue relative to containment bypass is a design-specific issue, and relates to how a particular designer addresses the functional SGTR requirements already provided in the requirements document.

In a letter dated September 17, 1992, Westinghouse indicated that the ALWR program developed in response to this issue includes Westinghouse input. The ALWR response to mitigate this issue includes the following elements: (1) both the evolutionary and passive PWR plants address SGTR containment bypass by features which significantly reduce the potential for core damage and for release directly to the atmosphere in SGTR accident sequences; (2) ALWR utilities are seeking design features in ALWR plants which could simplify operator response to SGTR; (3) evolutionary plants are designed to terminate SGTR by operator actions with a 30 minute grace period; and (4) passive plants have the same operator assisted capability but also include capability to mitigate SGTR without operator action.

The staff recommends that the Commission approve the position 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.

S. PRA Beyond Design Certification

A plant-specific PRA is an excellent method for assessing overall plant safety and integrating plant systems and human interactions. Careful review of a PRA can also reveal important engineering evaluations, assumptions, and uncertainties. In the design and design review processes, PRA insights can be used to select among design options, strengthen the design against previously known vulnerabilities, characterize the design, and evaluate the balance in the design between severe accident prevention and mitigation.

At the COL stage, the COL applicant will be able to provide site-specific information and detailed design information that was not available during the certification process. The COL applicant will be required to update the design-specific PRA to reflect site-specific information before COL issuance. During the construction stage, the COL holder will also be able to consider as-built information. 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 updated PRA insights, if properly evaluated and utilized, can strengthen programs and activities in areas such as training, emergency operating procedure development, reliability assurance, maintenance, and 10 CFR 50.59 evaluations. Therefore, the PRA should be revised to account for site-specific information, first-of-a-kind engineering developments, as-built (plant-specific) information refinements in the level of design detail, plant operational experience, and design changes. The COL applicant or COL holder should update the PRA to ensure that new information or design changes do not introduce new vulnerabilities or diminish the overall capability of the design to prevent and mitigate severe accidents. As plant experience data accumulates, failure rates (taken from generic data bases) and human errors assumed in the design PRA should be updated and incorporated as appropriate, into Operational Reliability Assurance Programs.

EPRI, Westinghouse, and NUMARC agree with the staff that a design-specific PRA has value and benefit, but they believe that the legal status of the PRA must be established under 10 CFR Part 52. Industry representatives expressed their desire to establish a common understanding of the legal and regulatory implications regarding the maintenance of the PRA. NUMARC, for example, stated that an ALWR design-specific PRA contains no unique or original design information that is not already reflected in associated SSAR Chapters. Accordingly, NUMARC believes that a design-specific PRA should be in neither Tier 1 nor Tier 2, but rather should be used as an analytical tool to assist the applicant and the NRC staff in evaluating the safety of the plant design.

In its letter of September 16, 1992, ACRS agreed that it is worthwhile for a plant operator to have an updated PRA. However, the ACRS was concerned about how the staff intends to use the updated PRA, how the staff thinks the licensee should use the updated PRA, and what should be required to update or keep the PRA current.

The staff is considering additional regulatory requirements to address revising the design-specific PRA after it has been completed and will discuss this issue in a future Commission paper regarding the form and content of a combined license.

T. 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 the 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's attention on the location and nature of the malfunction or disturbance. The extent to which this is achieved depends 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 that the power supplies of these systems are vulnerable to single failures. At present, the NRC has no requirements specific to the annunciator system. The acceptance criteria and guidelines for I&C systems (Appendix A to SRP, Section 7.1) developed from the GDC for the I&C, control room, and protection and reactivity control systems, do not specifically include the annunciator system. IEEE Standard 279, "Criteria for Protection Systems for Nuclear Power Generating Stations," states that 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.

When the operating U.S. plants were being designed, the international community generally observed the same requirements as those in the U.S. One of the few exceptions is 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 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 not part of the traditional safety 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

[1] For the purposes of this paper, the annunciator system is considered to consist of sets of alarms (which may be displayed on tiles, video display units (VDUs), or other devices) and sound equipment; logic and processing support; and functions to enable operators to silence, acknowledge, reset and test alarms.

systems. The staff discussed the need for such classification in some detail in SECY-91-292, "Digital Computer Systems for Advanced Light-Water Reactors," dated September 16, 1991.

The EPRI URDs for both evolutionary and passive ALWR plants states the following requirements:

> The main control room (MCR) shall contain compact, redundant operator workstations with multiple display and control devices that provide organized, hierarchical access to alarms, displays, and controls. Each workstation shall have the full capability to perform MCR functions as well as support division of tasks 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, and supervisory watchdog programs), as appropriate, to ensure that alarm, display, and control functions provided by the redundant workstations 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." This means that independence and separation must be provided between Class 1E and non-Class 1E circuits, even though the alarm, display, and control devices are to be located in a workstation. The alarm system is considered nonsafety-related, and, therefore, the nonsafety-grade alarm circuits must be separated from interfacing Class 1E circuits. The requirements for redundancy also apply to the power supplies associated with these work stations. 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 annunciators because of problems with their power supplies. The EPRI requirements for redundant workstations and displays that include the alarm functions are adequate for these stations.

The staff recommends that the Commission approve the position that the alarm system for ALWRs should meet the applicable 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.

III. Issues Limited to Passive Designs

A. Regulatory Treatment of Nonsafety Systems in Passive Designs

In contrast to both the current generation of LWR and the evolutionary ALWRs, the passive ALWR designs rely on safety systems that 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 contend that these designs do not include safety-grade ac electric power.

The passive ALWR designs also include nonsafety-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 AP600 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 that are needed to support these functions, including nonsafety standby diesel generators. In addition, the passive ALWR designs include nonsafety-grade active systems (such as the control room HVAC 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, "Quarterly Report on Emerging Technical Concerns," dated December 17, 1990, the staff identified the role of these nonsafety 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 a 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 (that is, pumped ECCSs

are more likely to overcome stuck valves). These uncertainties enhance the importance of the active nonsafety 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 an evaluation of not only the passive safety systems, but also the functional capability and availability of the active nonsafety systems to provide significant defense-in-depth and accident and core damage prevention capability.

For active systems that perform defense-in-depth functions, the EPRI requirements document for passive designs specifies requirements concerning performance and systems and equipment design. These include radiation shielding requirements (to permit access following an accident), redundancy, availability of nonsafety-grade electric power, and protection against internal hazards. The requirements also address safety analysis and testing to demonstrate system capability to satisfy defense-in-depth considerations. EPRI does not currently provide specific requirements for the reliability of these systems. However, in response to staff questions, EPRI has indicated that it is evaluating specific reliability targets and other measures to provide confidence that the passive plants will meet performance requirements. These requirements will address both passive safety and active nonsafety 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. This evaluation will determine which systems and components (including certain nonsafety systems) require the imposition of technical specifications, and the parameters of the technical specifications. The Reliability Assurance Program is expected to strongly influence these 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 AP600 and SBWR submittals. Significant decisions need to be made concerning the scope of staff review of the nonsafety systems and reliance on the passive systems. The staff will not require that the active systems meet all the safety-grade criteria, but there should be a high level of confidence that risk-significant active systems are designed in accordance with their performance/reliability missions to ensure their availability when needed.

The staff has held several meetings with EPRI to determine steps needed to resolve the issue of regulatory treatment of active nonsafety systems, and define the scope of requirements and acceptance criteria to ensure that they have adequate capability and availability when required. In a meeting between NRC and the Utility Steering Committee on January 22, 1993, an agreement was reached for an overall process for determining the regulatory treatment of nonsafety systems, and importance of passive systems and/or components for meeting NRC Safety Goals and requirements. On February 23, 1993, EPRI submitted a document containing a draft proposed process. This document is under staff review.

The staff is still evaluating this issue for the passive plant designs. The discussion provided in this section is to inform the Commission of the current status of the issue. The staff's proposed resolution of this issue will be provided to the Commission in a separate Commission paper, which will discuss the regulatory treatment of nonsafety systems in passive designs.

B. Definition of Passive Failure

Appendix A to 10 CFR Part 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 a component's capability to perform its intended safety functions. Fluid and electric systems are considered to be designed against an assumed single failure if the system maintains its ability to perform its safety functions in the event of a single failure of either any active component (assuming passive components function properly) or a passive component (assuming active components function properly). 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, "Single Failure Criterion," 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 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 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. Specifically, the staff imposed a passive failure in addition to the initiating event but not in others. 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 ensure 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 check valve failure as an active failure. This change appears necessary 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. In addition, 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. Redefining check valve failure in this manner would cause these valves to be evaluated in a more stringent manner than that used in previous licensing reviews.

The staff is still evaluating this issue for the passive plant designs. The staff's proposed resolution of this issue will be provided to the Commission in a separate Commission paper, which will discuss the regulatory treatment of active nonsafety systems in passive designs.

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, "Review of the Vendor's Test Programs to Support the Design Certification of Passive Light-Water Reactors," the staff determined that an early NRC assessment is needed. This assessment should address the vendor's analytical and experimental basis for demonstrating nuclear/thermal-hydraulic stability. In addition, it should identify any tests or analyses that may be needed to support the staff's technical evaluations of the issue. The NRC staff and its consultant, Oak Ridge National Laboratories (ORNL), have reviewed 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 and ORNL. These calculations showed that, while the system appears to be very stable under normal operating conditions, certain abnormal operating conditions might be reached under credible transient sequences. These abnormal conditions can result in the onset of density-wave power and flow oscillations. In addition, a low-flow and low-power instability caused by 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. Specifically, the staff concluded 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 representative of the divided chimney now employed. GE also 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. In addition, EPRI and GE believe that the SBWR is not vulnerable to a loop-type instability reported by the Japanese; rather, EPRI and GE contend that this instability was characteristic of the experimental apparatus used.

In SECY-92-339, "Evaluation of the General Electric Company's (GE's) Test Program to Support Design Certification for the Simplified Boiling Water Reactor (SBWR)," the staff noted that GE has modified the SBWR conceptual design. The staff also noted that GE has identified existing experimental data, which they believe constitutes 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 of the 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. 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 considers this a technical issue and expects to resolve this issue with GE through its normal review of the SBWR design certification application and through its review of GE's SBWR testing program. The staff will interact with the Commission if additional policy issues are identified during its review of the SBWR application or vendor SBWR testing program.

D. Safe Shutdown Requirements

GDC 34 of Appendix A to 10 CFR Part 50 requires that a residual heat removal system be provided to remove residual heat from the reactor core so that specified acceptable fuel design limits (SAFDLs) are not exceeded. RG 1.139 and Branch Technical Position (BTP) 5-1 implement this requirement and set forth conditions for cold shutdown 93.3 °C (200 °F) for a PWR and 100 °C (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.

Because passive ALWR designs use passive heat removal systems for decay heat removal, they are limited by the inherent ability of the passive heat removal processes. These designs 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 of the AP600 or the isolation condenser of the SBWR. Even though active shutdown cooling systems are available to bring the reactor to cold-shutdown or refueling conditions, these active RHR systems are not safety-grade and do not comply with the guidance of RG 1.139 or BTP 5-1.

EPRI states that it is not necessary for passive safety systems to be capable of achieving cold shutdown. EPRI bases this contention on the belief that the passive decay heat removal (DHR) systems have an inherently 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 215.6 °C (420 °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. Also, operation of the passive DHR system does not require any ac power or pumps. EPRI further states that the nonsafety 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, the staff must resolve several issues before reaching a final position on this matter. These issues include reliability criteria for the nonsafety systems which have the capability to bring the plant to cold shutdown and the acceptability of 215.6 °C (420 °F) as a safe, long-term state. The staff's proposed resolution of this issue will be provided to the Commission in a separate Commission paper, which will discuss the regulatory treatment of nonsafety systems in passive designs.

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 to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. In current plants, safety-grade filtered control

room HVAC systems with charcoal absorbers are used to ensure that radiation doses to operators could be maintained within the GDC 19 criteria in the event of an accident.

In SRP Section 6.4, the staff defined the acceptable operator dose criterion in terms of specific whole body and organ doses (5 rem to the whole body, and 30 rem each to the thyroid and skin). Recently, the NRC 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 include use of the effective dose equivalent, did not change the dose criteria for control room operators to conform with GDC 19.

Originally, EPRI proposed an exposure limit for control room operators of 5 rem whole body, 75 rem skin, and 300 rem thyroid. EPRI stated that each operator would be provided with individual breathing apparatus and protective clothing, if required, to meet regulatory limits. The staff determined that EPRI's requirements for the thyroid and beta skin doses were not adequately justified. The staff indicated to EPRI that the long-term use of a breathing apparatus during design-basis accidents has never been allowed. More importantly, the long-term use of a breathing apparatus is likely to degrade operator performance during and following an accident.

EPRI also stated that the control room would be designed to be maintained during a 72-hour period as the primary location from which personnel can safely operate in the event of an accident. It was and is the staff's position that, depending upon the accident, the required duration may be much longer than 72 hours. GDC 19 states that "adequate radiation protection shall be provided to permit access to and occupancy of the control room under accident conditions...for the duration of the accident." Consequently, the staff concluded that analyses of control room habitability should consider the duration of the accident (which may extend well beyond the EPRI-proposed 72-hour period) as the design basis.

In order to resolve this matter, the staff proposed that EPRI and the vendors provide a high level of assurance that the control room ventilation system will be available when needed. Because the system may not need to meet all of the safety-grade criteria, it may be appropriate to allow some credit for nonsafety-grade 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 nonsafety systems. It should be noted that unlike the case of core cooling, there is no passive safety-grade system for defense-in-depth of control room habitability.

In its letter of May 5, 1992, EPRI proposed an alternative which would use a safety-grade pressurization system capable of being recharged remotely after 72 hours. In that enclosure, EPRI stated that Volume III of the Utility Requirements Document would be revised to require: (1) a passive, safety-grade control room pressurization system which would use bottled air to keep operator doses within the limits of GDC 19 and SRP Section 6.4, Revision 2 for

the first 72 hours of the event, and (2) safety-grade connections for the pressurization system to allow use of offsite, portable air supplies if needed after 72 hours to minimize operator doses. The staff agrees with EPRI's commitment to limit the operator doses to those specified in GDC 19 and SRP Section 6.4, Revision 2. However, the staff is still evaluating the proposal to utilize a safety-grade pressurization system and has serious reservations concerning the feasibility and the capability of a pressurization system to maintain the control room habitability.

In its letter of August 17, 1992, ACRS indicated that they had discussed the subject of control room habitability with EPRI and the staff during a meeting on June 4 and 5, 1992. At that meeting, the staff told ACRS that they were evaluating the EPRI proposal utilizing the safety-grade pressurization system. ACRS indicated that they had several comments regarding the design features of the passive control room pressurization system proposed by EPRI. ACRS stated that the staff should take these comments into account in performing its evaluation, and that ACRS may provide additional recommendations after the staff has completed its evaluation.

The staff is currently reviewing the new severe accident source term proposed by EPRI in conjunction with the staff's technical update of 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. Specifically, this outcome will include 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 the control building that houses the control room. Therefore, the staff is unable to complete its control room habitability assessment until issues concerning the source term and its behavior mechanism are satisfactorily resolved.

The staff plans to present its proposed resolution of this issue in a separate Commission paper, which will discuss issues related to source term.

F. Radionuclide Attenuation

EPRI and the passive ALWR designers rely on assumptions involving fission product removal inside containment by natural removal effects and holdup by the secondary building and piping systems. A containment spray system is not mandated by the EPRI requirements document for passive plant designs. The staff is concerned about the uncertainty in quantifying the holdup phenomena in the auxiliary building and that use of the auxiliary building for holdup may require imposition of additional restrictions on the auxiliary building during normal operation, with which the licensee may have difficulty complying.

This issue also affects the control room habitability issue discussed in Section III.E of this report. The relationship arises because the industry indicates that fission products will be removed before they reach the control room air intake 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 fission product attenuation in the main steamlines and condenser is appropriate for the passive BWR design. This question arises because the main steamlines downstream of the main steam isolation valves and associated condenser are not designed to withstand the 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 deposition rates varying with temperature, pressure, gas composition, surface material, and particulate size.

The staff's proposed resolution of this issue will be presented in a separate Commission paper, which will discuss source term related issues.

G. Simplification of Offsite Emergency Planning

EPRI has proposed to significantly simplify offsite 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 offsite dose. EPRI's proposal would eliminate requirements for early notification of the public, detailed evacuation planning, and provisions for exercising the offsite plan. The onsite emergency plan and limited offsite actions would be retained. EPRI has identified this matter as a plant optimization subject.

During a meeting with the staff on January 30, 1992, EPRI proposed to work with the staff to define a process for addressing simplification of emergency planning. This simplification would include developing technical criteria and methods that, if met, would justify such action. It would also include defining the process for implementing 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 to the emergency planning requirements of 10 CFR Part 50 and 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 required for the plume exposure pathway emergency planning zone. Any decision on emergency planning requirements for the passive design should reflect a plant's ability to prevent the significant release of

radioactive material or to provide very long delay times before a release for all but the most unlikely events. Before relaxing emergency planning requirements, the staff will require a high degree of assurance that all potential containment bypass accident sequences have a very low likelihood. 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, as a policy matter, it may be that even very low calculated probability values should not be considered a sufficient basis for changes to emergency planning requirements.

The staff will evaluate this issue for the passive plant designs when sufficient supporting information becomes available. The staff plans to update the status of this review in a separate Commission paper.

H. Role of the Passive Plant Control Room Operator

In SECY-91-272, "Role of Personnel and Advanced Control Rooms in Future Nuclear Power Plants," dated August 27, 1991, the staff discussed 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. 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 safely operate a passive plant, 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 evolutionary plants. These new functions and tasks will be associated with the new operational philosophy noted above, the increase in automation, and the greater use of advanced technology in 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 process 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, as specified within Chapter 10 of the EPRI URD. EPRI also requires a full scope control room design simulator for this testing. To correct the problem of insufficient focus on the operator in previous designs, EPRI indicated that this requirement should adequately ensure that the human component in the man/machine interface system is explicitly included. However, EPRI maintains that the difference in the role of the operator in a passive plant control room is limited to the details and timing

of the actions performed. The staff will consider the operator's role in its review of each passive plant applicant's control room design under 10 CFR Part 52.

The staff concludes that an extensive man-in-the-loop test and evaluation program will be necessary for the passive plant control room designs. This testing will address the extent of differences in the operator's role in a passive plant control room since it will simulate tasks necessary for maintaining plant safety following an event. 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.

In its letter of August 21, 1992, EPRI restated that the approach to operating passive designs is the same as for evolutionary or existing designs. The staff does not agree with this position. However, the EPRI ALWR Program has provided for man-in-the-loop testing and a full scope control room design simulator to ensure that the human component is explicitly considered and is acceptable. The staff has continued discussions with EPRI and passive plant vendors and believes this approach will resolve the differences in the position on operating philosophy.

In its letter of September 16, 1992, ACRS agreed with the staff that sufficient man-in-the-loop testing and evaluation should demonstrate that the operator's functions and tasks are properly integrated into the man/machine interface design.

Therefore, the staff recommends that the Commission approve the position that sufficient man-in-the-loop testing and evaluation must be performed. In addition, a fully functional integrated control room prototype is likely to be necessary for passive plant control room designs to demonstrate that functions and tasks are properly integrated into the man/machine interface design.

These requirements will be incorporated into the DAC. Each applicant may provide justification that a control room prototype of reduced scope is sufficient to ensure that functions and tasks are properly integrated in the man/machine interface design.

## ALWR ISSUES CROSS-REFERENCE MATRIX

| Category | Issue | Title | Commission Papers |
|---|---|---|---|
| I. SECY-90-016 Issues | A. | Use of Physically Based Source Term | 86-228<br>88-203<br>89-013<br>89-153<br>89-228<br>89-341<br>90-016<br>90-307<br>90-329<br>90-341<br>90-353<br>92-127<br>Draft-1[1] |
| | B. | ATWS | 89-153<br>89-228<br>90-016<br>90-353<br>Draft-1 |
| | C. | Mid-Loop Operation | 89-228<br>90-016<br>90-353<br>Draft-1 |
| | D. | Station Blackout | 89-013<br>89-153<br>89-228<br>90-016<br>90-329<br>90-353<br>Draft-1 |
| | E. | Fire Protection | 89-013<br>89-228<br>90-016<br>90-353<br>Draft-1 |

[1]"Draft-1" 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.

| Category | Issue | Title | Commission Papers |
|---|---|---|---|
| I. SECY-90-016 Issues (cont.) | F. | Intersystem LOCA | 89-153<br>89-228<br>90-016<br>90-353<br>Draft-1 |
| | G. | Hydrogen Control | 89-013<br>89-153<br>89-228<br>90-016<br>90-329<br>90-353<br>Draft-1 |
| | H. | Core Debris Coolability | 89-153<br>89-228<br>90-016<br>90-353<br>92-092<br>Draft-1 |
| | I. | High-Pressure Core Melt Ejection | 89-228<br>90-016<br>90-353<br>92-092<br>Draft-1 |
| | J. | Containment Performance | 89-228<br>90-016<br>90-353<br>91-273<br>92-092<br>Draft-1 |
| | K. | Dedicated Containment Vent Penetration | 89-153<br>89-228<br>90-016<br>90-329<br>90-353<br>92-092<br>Draft-1 |
| | L. | Equipment Survivability | 89-228<br>90-016<br>90-353<br>Draft-1 |
| | M. | Elimination of OBE | 89-013<br>90-016<br>90-329<br>90-353 |

| Category | Issue | Title | Commission Papers |
|---|---|---|---|
| I. SECY-90-016 Issues (cont.) | M. | Elimination of OBE (cont.) | 91-135<br>Draft-1<br>Draft-2[2] |
| | N. | In-Service Testing of Pumps and Valves | 89-228<br>90-016<br>90-353<br>91-273<br>Draft-1 |
| II. Other Evolutionary and Passive Design Issues | A. | Industry Codes and Standards | 91-273<br>Draft-1 |
| | B. | Electrical Distribution | 91-078<br>Draft-1 |
| | C. | Seismic Hazard Curves and Design Parameters | 91-135<br>Draft-1 |
| | D. | Leak-Before-Break | 89-013<br>Draft-1 |
| | E. | Classification of Main Steamlines in BWRs | Draft-1 |
| | F. | Tornado Design Basis | Draft-1 |
| | G. | Containment Bypass | Draft-1 |
| | H. | Containment Leak Rate Testing | 89-013<br>89-228<br>91-273<br>Draft-1 |
| | I. | Post-Accident Sampling System | Draft-1 |
| | J. | Level of Detail | 90-241<br>90-377<br>Draft-1 |
| | K. | Prototyping | 91-074<br>91-273<br>Draft-1 |

[2]"Draft-2" refers to the draft Commission paper, "Design Certification and Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light-Water Reactor Designs," which was forwarded to the Commission on June 25, 1992, and made available to the public on July 1, 1992.

| Category | Issue | Title | Commission Papers |
|---|---|---|---|
| II. Other Evolutionary and Passive Design Issues (cont.) | L. | ITAAC | 91-178<br>91-210<br>92-053<br>92-196<br>92-214<br>92-287<br>92-294<br>Draft-1 |
| | M. | Reliability Assurance Program | 89-013<br>92-133<br>Draft-1 |
| | N. | Site-Specific PRAs and Analyses of External Events | 89-013<br>Draft-1<br>Draft-2 |
| | O. | SAMDAs | 91-229<br>Draft-1 |
| | P. | Generic Rulemaking Related to Design Certification | 91-262<br>Draft-1 |
| | Q. | Defense Against Common-Mode Failures in Digital I&C Control Systems | 91-292<br>Draft-2 |
| | R. | Multiple SG Tube Ruptures | 92-133<br>Draft-2 |
| | S. | PRA Beyond Design Certification | Draft-2 |
| | T. | Control Room Annunciator Reliability | Draft-2 |
| III. Issues Limited to Passive Design | A. | Regulatory Treatment of Active Nonsafety Systems | 89-013<br>90-406<br>92-133<br>Draft-1<br>Draft-2 |
| | B. | Definition of Passive Failure | 77-439<br>Draft-1 |
| | C. | Thermal-Hydraulic Stability of the SBWR | 89-153<br>91-273<br>92-339<br>Draft-1 |

| Category | Issue | Title | Commission Papers |
|---|---|---|---|
| III. Issues Limited to Passive Design (Cont.) | D. | Safe Shutdown Requirements | Draft-1<br>92-133 |
| | E. | Control Room Habitability | 92-133<br>Draft-1<br>Draft-2 |
| | F. | Radionuclide Attenuation | 92-127<br>92-133<br>Draft-1 |
| | G. | Simplification of Offsite Emergency Planning | 88-203<br>Draft-1 |
| | H. | Role of the Passive Plant Control Room Operator | 91-272<br>Draft-2 |

## COMMISSION PAPERS APPLICABLE TO ALWRs

SECY-77-439, "Single Failure Criterion," August 17, 1977.

SECY-86-228, "Introduction of Realistic Source-Term Estimates into Licensing," August 6, 1986.

SECY-88-147, "Integration Plan for Closure of Severe Accident Issues," May 25, 1988.

SECY-88-202, "Standardization of Advanced Reactor Designs," July 15, 1988.

SECY-88-203, "Key Licensing Issues Associated with DOE-Sponsored Advanced Reactor Designs," July 15, 1988.

SECY-89-012, "Staff Plans for Accident Management Regulatory and Research Programs," January 18, 1989.

SECY-89-013, "Design Requirements Related to the Evolutionary Advanced Light-Water Reactors (ALWRs)," January 19, 1989.

SECY-89-153, "Severe Accident Design Features of the Advanced Boiling-Water Reactor (ABWR)," May 10, 1989.

SECY-89-178, "Policy Statement Integration," June 9, 1989.

SECY-89-228, "Draft Safety Evaluation Report on Chapter 5 of the Advanced Light-Water Reactor Requirements Document," July 28, 1989.

SECY-89-341, "Updated Light-Water Reactor (LWR) Source-Term Methodology and Potential Regulatory Applications," November 6, 1989.

SECY-90-016, "Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," January 12, 1990.

SECY-90-065, "Evolutionary and Passive Advanced Light Water Reactor Resources and Schedules," March 7, 1990.

SECY-90-146, "Process, Schedule, and Resources for the Review of Evolutionary and Passive Advanced Light-Water Reactors," April 20, 1990.

SECY-90-241, "Level of Detail Required for Design Certification Under Part 52," July 11, 1990.

SECY-90-307, "Impacts of Source-Term Timing on NRC Regulatory Positions," August 30, 1990.

SECY-90-313, "Status of Accident Management Program and Plans for Implementation," September 5, 1990.

SECY-90-329, "Comparison of the General Electric Advanced Boiling-Water Reactor (ABWR) Design and the Electric Power Research Institute's (EPRI's) Advanced Light-Water Reactor (ALWR) Requirements Document," September 20, 1990.

SECY-90-341, "Staff Study on Source-Term Update and Decoupling Siting from Design," October 4, 1990.

SECY-90-353, "Licensing Review Basis for the Combustion Engineering, Inc. System 80+ Evolutionary Light-Water Reactor," October 12, 1990.

SECY-90-377, "Requirements for Design Certification Under 10 CFR Part 52," November 8, 1990.

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.

SECY-91-135, "Conclusions of the Probabilistic Seismic Hazard Studies Conducted for Nuclear Power Plants in the Eastern United States," May 14, 1991.

SECY-91-161, "Schedules for the Advanced Reactor Reviews and Regulatory Guidance Revisions," May 31, 1991.

SECY-91-178, "Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Design Certifications and Combined Licenses," June 12, 1991.

SECY-91-210, "Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Requirements for Design Review and Issuance of a Final Design Approval," July 16, 1991.

SECY-91-229, "Severe Accident Mitigation Design Alternatives for Certified Standard Designs," July 31, 1991.

SECY-91-239, "Preapplication Reviews of Advanced LWR Designs," August 5, 1991.

SECY-91-262, "Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light-Water Reactor (LWR) Designs," August 16, 1991.

SECY-91-272, "Role of Personnel and Advanced Control Rooms in Future Nuclear Power Plants," August 27, 1991.

SECY-91-273, "Review of Vendors' Test Programs to Support the Design Certification of Passive Light Water Reactors," August 27, 1991.

SECY-91-292, "Digital Computer Systems for Advanced Light-Water Reactors," September 16, 1991.

SECY-91-348, "Issuance of Final Revision to Appendix J to 10 CFR 50, and Related Final Regulatory Guide 1.XXX (MS 021-5)," October 25, 1991.

SECY-92-030, "Integral System Testing Requirements for Westinghouse's AP600 Plant," January 27, 1992.

SECY-92-037, "Need for NRC-Sponsored Confirmatory Integral System Testing of the Westinghouse AP600 Design," January 31, 1992.

SECY-92-053, "Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews," February 19, 1992.

SECY-92-092, "The Containment Performance Goal, External Event Sequences, and the Definition of Containment Failure for Advanced Light-Water Reactors," March 17, 1992.

SECY-92-120, "NRC Staff Review Schedules for the Westinghouse AP600 and the General Electric (GE) Simplified Boiling-Water Reactor (SBWR) Designs," April 7, 1992.

SECY-92-127, "Revised Accident Source Terms for Light-Water Nuclear Power Plants," April 10, 1992.

SECY-92-133, "Draft Safety Evaluation Report for Volume I and Volume III of the Electric Power Research Institute's Advanced Light-Water Reactor Requirements Document," April 14, 1992.

SECY-92-134, "NRC Construction Inspection Program for Evolutionary and Advanced Reactors Under 10 CFR Part 52," April 15, 1992.

SECY-92-170, "Rulemaking Procedures for Design Certification," May 8, 1992.

SECY-92-196, "Development of Design Acceptance Criteria (DAC) for the Advanced Boiling-Water Reactor (ABWR)," May 28, 1992.

"Issues Pertaining to Evolutionary and Passive Light-Water Reactors and Their Relationship to Current Regulatory Requirements," draft Commission paper forwarded to the Commission on February 20, 1992, and made available to the public on February 27, 1992.

SECY-92-211, "NRC Confirmatory Integral System Testing for the General Electric SBWR Design," June 5, 1992.

SECY-92-214, "Development of Inspections, Test, Analyses, and Acceptance Criteria (ITAAC) for Design Certifications," June 11, 1992.

SECY-92-219, "NRC-Sponsored Confirmatory Testing of the Westinghouse AP600 Design," June 16, 1992.

"Design Certification and Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light-Water Reactor Designs," draft Commission paper forwarded to the Commission on June 25, 1992, and made available to the public on July 1, 1992.

SECY-92-277, "Final Safety Evaluation Report for Volume II of the Electric Power Research Institute's Advanced Light Water Reactor Requirements Document," August 10, 1992.

SECY-92-287, "Form and Content for a Design Certification Rule," August 18, 1992.

SECY-92-292, "Advance Notice of Proposed Rulemaking on Severe Accident Plant Performance Criteria for Future LWRs," August 21, 1992.

SECY-92-294, "Acceptance Review of the Westinghouse Electric Corporation's Application for Final Design Approval and Design Certification for the AP600 Design," August 24, 1992.

SECY-92-299, "Development of Design Acceptance Criteria (DAC) for the Advanced Boiling-Water Reactor (ABWR) in the Areas of Instrumentation and Controls (I&C) and Control Room Design," August 27, 1992.

SECY-92-327, "Reviews of Inspections, Test, Analyses, and Acceptance Criteria (ITAAC) for the General Electric (GE) Advanced Boiling Water Reactor (ABWR)," September 22, 1992.

SECY-92-339, "Evaluation of the General Electric Company's (GE's) Test Program to Support Design Certification for the Simplified Boiling-Water Reactor," October 6, 1992.

SECY-92-368, "Final Rule Amending 10 CFR Part 52," October 29, 1992.

SECY-92-381, "Rulemaking Procedures for Design Certification," November 10, 1992.

SECY-92-403, "Acceptance Review of GE Nuclear Energy's (GE's) Application for Final Design Approval (FDA) and Design Certification (DC) of the Simplified Boiling-Water Reactor Design (SBWR)," December 3, 1992.

STAFF REQUIREMENTS MEMORANDUM
"SECY-93-087 - POLICY, TECHNICAL,
AND LICENSING ISSUES PERTAINING TO
EVOLUTIONARY AND ADVANCED LIGHT-WATER
REACTOR (ALWR) DESIGNS"
JULY 21, 1993

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICE OF THE
SECRETARY

July 21, 1993

MEMORANDUM FOR: James M. Taylor, Executive Director for Operations

FROM: Samuel J. Chilk, Secretary

SUBJECT: **SECY-93-087 - POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS**

This is to advise you that the Commission (with all Commissioners agreeing) has approved the items for which the staff requested a decision as follows:

1. I.E. Fire Protection:

   The Commission approves the staff's position that the passive plants should also be reviewed against the enhanced fire protection criteria approved in the Commission's SRM of June 26, 1990.

   The Commission would like to be kept informed of the staff's resolution of the issue related to common-mode failures through common ventilation systems.

2. I.F. Intersystem Loss-of-Coolant Accident:

   The Commission approves the staff's position that the passive plants should also be reviewed for compliance with the intersystem LOCA criteria approved in the Commission's SRM of June 26, 1990.

   The staff should clarify the intent of the phrase "could not practically be designed to meet such a criterion."

3. I.G. Hydrogen Control:

   The Commission approves the staff's position that the passive plants should be designed, as a minimum, to the same requirements applied to evolutionary designs. Specifically, passive plants must:

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SECY NOTE: This SRM and the vote sheets of all Commissioners will be made publicly available in 10 working days from the date of this SRM. (SECY-93-087 was previously released to the public).

- o accommodate hydrogen generation equivalent to a 100% metal-water reaction of the fuel cladding;

- o limit containment hydrogen concentration to no greater than 10%; and

- o provide containment-wide hydrogen control (such as igniters or inerting) for severe accidents.

The Commission approves the staff's clarification, as expressed at the Commission briefing, that the possible use of passive autocatalytic hydrogen recombiners should not be precluded from consideration a priori. The staff is cautioned to consider carefully the relatively slow time response of autocatalytic recombiners as a possible impediment to their efficiency.

The staff's resolution of the issue of the placement of hydrogen control devices should be provided to the ACRS and the Commission for information.

4. I.H. Core Debris Coolability:

The Commission approves the staff's position that both the evolutionary and passive ALWR designs meet the following criteria:

- o Provide reactor cavity floor space to enhance debris spreading.

- o Provide a means to flood the reactor cavity to assist in the cooling process.

- o Protect the containment liner and other structural members with concrete, if necessary.

- o Ensure that the best estimate environmental conditions (pressure and temperature) resulting from core-concrete interactions do not exceed Service Level C for steel containments or Reactor Load Category for concrete containments, for approximately 24 hours. Ensure that the containment capability has margin to accommodate uncertainties in the environmental conditions from core-concrete interactions.

With regard to the $0.02 m^2/MWt$ reactor vessel cavity floor area, the staff should continue its research activities and supporting analyses, as documented in its May 19, 1993 letter to the ACRS.

With respect to the containment response to ex-reactor vessel core debris, the staff should not limit licensees to

only one method for addressing containment responses to severe accident events but also permit other technically justified means for demonstrating adequate containment response.

5. I.I. High Pressure Core Melt Ejection:

The Commission approves the staff's position for the general criteria that the evolutionary and passive LWR designs:

- o provide a reliable depressurization system; and
- o provide cavity design features to decrease the amount of ejected core debris that reaches the upper containment.

6. I.J. Containment Performance:

The recommendations on containment performance, as outlined in SECY 93-087, could be read to imply that the staff is no longer proposing to use the concept of conditional containment failure probabilities (CCFP). However, based on discussions held during the Commission meeting on this subject, the staff informed the Commission that it intends to continue to apply the 0.1 CCFP in implementing the Commission's defense in depth regulatory philosophy and the Commission's policy on Safety Goals.

Therefore, the Commission approves the staff's position to use the following deterministic containment performance goal in the evaluation of the passive ALWRs as a complement to the CCFP approach approved by the Commission in its SRM of June 26, 1990:

> "The containment should maintain its role as a reliable, leak-tight barrier (for example, by ensuring that containments stresses do not exceed ASME Service Level C limits for metal containments, or Factored Load Category for concrete containments) for approximately 24 hours following the onset of core damage under the more likely severe accident challenges and, following this period, the containment should continue to provide a barrier against the uncontrolled release of fission products."

The Commission approves the staff's interim approach subject to the staff's review and recommendations resulting from public comments on the "Advance Notice of Proposed Rulemaking on Severe Accident Plant Performance Criteria for Future ALWRs."

7. **I.K. Dedicated Containment Vent Penetration:**

The Commission approves the staff's position that the need for a containment vent for the passive plant designs should be evaluated on a design-specific basis.

8. **I.L. Equipment Survivability:**

The Commission approves the staff's position that the passive plant design features provided only for severe-accident mitigation need not be subject to the environmental qualification requirements of 10 CFR Section 50.49; quality assurance requirements of 10 CFR Part 50, Appendix B; and redundancy/diversity requirements of 10 CFR Part 50, Appendix A.

9. **I.M. Elimination of Operating-Basis Earthquake (OBE):**

The Commission approves the staff's recommendation to account for earthquake cycles in the fatigue analyses of piping systems performed until the new guidance is issued, using two SSE events with 10 maximum stress cycles per event (20 full cycles of the maximum SSE stress range). Alternatively, the number of fractional vibratory cycles equivalent to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Appendix D of IEEE Standard 344-1987.

The Commission approves the staff's recommendation that the effects of anchor displacements in the piping caused by an SSE be considered with the Service Level D limit.

The Commission approves the staff's recommendation to eliminate the OBE from the design of systems, structures, and components. When the OBE is eliminated from the design, no replacement earthquake loading should be used to establish the postulated pipe rupture and leakage crack locations.

The Commission approves the staff's recommendation that the mechanistic pipe break and high-energy leakage crack locations determined by the piping high stress (without the OBE) and fatigue locations may be used for equipment environmental qualification and compartment pressurization purposes.

The Commission agrees that with the elimination of the OBE, two alternatives exist that will essentially maintain the requirements provided in IEEE Standard 344-1987 to qualify equipment with the equivalent of five OBE events followed by

one SSE event (with 10 maximum stress cycles per event). Of these alternatives, the equipment should be qualified with five one-half SSE events followed by one full SSE event. Alternatively, a number of fractional peak cycles equivalent to the maximum peak cycles for five one-half SSE events may be used in accordance with Appendix D of IEEE Standard 344-1987 when followed by one full SSE.

The Commission agrees that the above requirements should also apply to passive ALWRs.

The Commission understands that the OBE will continue to be used as a threshold criterion for conducting inspections following an earthquake event. The staff should keep the Commission and the ACRS informed as the staff's further analysis and review proceed.

10. I.N. Inservice Testing of Pumps and Valves:

The Commission has no objection to the staff's position, but understands that further elaboration on this issue will be forthcoming from the staff.

11. II.A. Industry Codes and Standards:

The Commission approves the staff's position that consistent with past practice, that staff will review both evolutionary and passive plant design applications using the newest codes and standards that have been endorsed by the NRC. Unapproved revisions to codes and standards will be reviewed on a case-by-case basis.

12. II.D. Leak Before Break:

The Commission approves the staff's recommendation that the leak before break approach should be applied to both the evolutionary and the passive ALWRs seeking design certification under 10 CFR Part 52. This approval should be limited to instances in which appropriate bounding limits are established using preliminary analysis results during the design certification phase and verified during the COL phase by performing the appropriate ITAAC.

13. II.E. Classification of Main Steamlines in Boiling Water Reactors:

The Commission approves the staff's position that neither the main steam drain and bypass lines from the first valve up to the condenser inlet, nor the piping between the turbine stop valve and the turbine inlet should be classified as safety-related or as seismic Category I. Rather, these lines should be analyzed using a dynamic

seismic analysis to demonstrate structural integrity under SSE loading conditions. The turbine stop, control, and bypass valves and the main steam lines from the turbine control valves to the turbine shall meet all of the quality group and quality assurance guidelines specified in SRP Section 3.2.2, Appendix A. Further, that seismic analyses be performed to ensure that the condenser anchorages and the piping inlet nozzle to the condenser are capable of maintaining their structural integrity during and after the SSE.

The Commission approves the above-described approach to resolve the main steamline classification for both evolutionary and passive ALWRs.

14. II.F. Tornado Design Basis:

The Commission approves the staff's position that a maximum tornado wind speed of 482 km/hr (300 mph) be used in the design-basis tornado employed in the design of evolutionary and passive ALWRs.

15. II.H. Containment Leak Rate Testing:

The Commission approves the staff's position that until the rule change proceedings for Appendix J of 10 CFR Part 50 are completed, the maximum interval between Type C leakage rate tests for both evolutionary and passive plant designs should be 30 months, rather than the 24 months maximum interval currently required in Appendix J to 10 CFR Part 50.

16. II.I. Post-Accident Sampling System (PASS):

The Commission approves the staff's position as modified below.

The Commission approves the staff's recommendation that the post-accident sampling systems for evolutionary and passive ALWRs of the pressurized water reactor type be required to have the capability to ~~analyze~~ determine the gross amount of dissolved gases (not necessarily a pressurized sample) ~~and chloride in accordance with the requirements~~ as an acceptable means of satisfying the intent of 10 CFR 50.34(f)(2)(viii) and Item II.B.3 of NUREG-0737.

The Commission agrees that the time for taking these samples can be extended to 24 hours following the accident.

The Commission agrees that for evolutionary and passive ALWRs of the boiling water reactor type, there would be no need for the post-accident sampling system to analyze dissolved gases.

The Commission approves the deviation from the requirements of Item II.B.3. of NUREG-0737 with regard to requirements for sampling reactor coolant for boron concentration and radioactivity measurements using the post-accident sampling system in evolutionary and passive ALWRs. The modified requirement would require the capability to take boron concentration samples and radioactivity measurements 8 hours and 24 hours, respectively, following the accident.

The Commission approval is based on the fact that the PASS system is an existing requirement and on the belief that a relatively simple system can be designed to meet the modified requirement. It is the Commission's understanding that a system can be designed which is simple, does not require chemical analysis of the gases in solution, and will provide the reactor operator information as to whether significant amounts of non-condensible gases exist in the reactor coolant.

17. <u>II.N. Site-Specific Probabilistic Risk Assessments and Analysis of External Events</u>

The Commission approves, in part, and disapproves, in part, the staff's position on site-specific probabilistic risk assessment and analysis of external events, as listed below.

The Commission approves the position that the analyses submitted in accordance with 10 CFR 52.47 should include an assessment of internal and external events.

The Commission disapproves the staff's recommendation to use two times the Design Basis SSE for margins-type assessment of seismic events.

The Commission approves the use of 1.67 times the Design Basis SSE for a margin-type assessment of seismic events.

The Commission approves the following staff recommendation, as modified:

> PRA insights will be used to support a margins-type assessment of seismic events. A PRA-based seismic margins analysis will consider sequence-level High Confidence, Low Probability of Failures (HCLPFs) and fragilities for all sequences leading to core damage or containment failures up to approximately <u>one and two-thirds the ground motion acceleration of the Design Basis SSE.</u>

The Commission approves the staff's position that the simplified probabilistic methods, such as <u>but not limited to</u> EPRI's FIVE methodology, will be used to evaluate fires.

The Commission approves the staff's position that traditional probabilistic techniques should be used to evaluate internal floods.

The Commission approves the staff's position that the ALWR vendors should perform bounding analyses of site-specific external events likely to be a challenge to the plant (such as river flooding, storm surge, tsunami, volcanism, high winds, and hurricanes).

The Commission approves the staff's position that when a site is chosen, its characteristics should be compared to those assumed in the bounding analyses to ensure that the site is enveloped.

The Commission approves the staff's position that if the site is enveloped, the COL applicant need not perform further PRA evaluations for these external events. The COL applicant should perform site-specific PRA evaluations to address any site-specific hazards for which a bounding analysis was not performed or which are not enveloped by the bounding analyses to ensure that no vulnerabilities due to siting exist.

18. II.Q. Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems:

The Commission approves, in part, and disapproves, in part, the staff's recommendation. The Commission has approved a revised position, as follows:

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 adequately been addressed.

2. In performing the assessment, 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) using best-estimate methods. 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 function. The diverse or different function may be performed by a non-safety system if the system

is of sufficient quality to perform the necessary function under the associated event conditions.

4. A set of ~~safety-grade~~ displays and controls located in the main control room shall be provided for manual, system-level actuation of critical safety functions and monitoring of parameters that support the safety functions. The displays and controls shall be independent and diverse from the safety computer system identified in items 1 and 3 above.

The staff's position has been modified in essentially two respects:

First, inasmuch as common mode failures are beyond design-basis events, the analysis of such events should be on a best-estimate basis.

Second, the staff indicates in its discussion of the third part of its position that "The diverse or different function may be performed by a non-safety system if the system is of sufficient quality to perform the necessary function under the associated event conditions." Therefore, this clarification has been added to the fourth part of the staff's position (which refers to a subset of the safety functions referred to in the third part) by removing the safety grade requirement. Further, the remainder of the discussion under the fourth part of the staff position is highly prescriptive and detailed (e.g., "shall be evaluated," "shall be sufficient," shall be hardwired," etc.). The Commission approves only that such prescriptiveness be considered as general guidance, the practicality of which should be determined on a case-by-case basis.

19. **II.R. Steam Generator Tube Ruptures:**

**II.R.1. Multiple Steam Generator Tube Ruptures:**

The Commission approves the staff's position to require that analysis of multiple steam generator tube ruptures (STGRs) involving two to five steam generator tubes be included in the application for design certification for the passive PWRs. The Commission understands that, as discussed in the Commission meeting on this SECY paper, since the steam generator multi-tube rupture event is beyond the design basis requirements for PWRs, realistic or best-estimate analytical assumptions may be used to assess plant responses.

II.R.2. Containment Bypass Potential Resulting From SGTRs:

The Commission approves the staff's recommendation 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 steam generator tube ruptures.

20. II.T. Control Room Annunciator Alarm) Reliability:

The Commission approves the staff's recommendation that the alarm system for ALWRs should meet the applicable EPRI requirements 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.

21. III.H. Role of the Passive Plant Control Room Operator:

The Commission approves the staff's recommendation that sufficient man-in-the-loop testing and evaluation must be performed. In addition, a fully functional integrated control room prototype is likely to be necessary for passive plant control room designs to demonstrate that functions and tasks are properly integrated into the man/machine interface.

Finally, the staff and industry should meet to ensure a common understanding of the requirements such that industry's design activities are appropriately directed to comply with the requirements.

The Commission commends the staff for a job well done on the highly complex technical issues presented in this paper.

cc: The Chairman
Commissioner Rogers
Commissioner Remick
Commissioner de Planque
OGC
IG
ACRS
OCA
OPA

ANNEX B

DOCUMENTS RELATED TO
SECY-94-084
"POLICY AND TECHNICAL ISSUES ASSOCIATED
WITH THE REGULATORY TREATMENT OF
NON-SAFETY SYSTEMS IN PASSIVE PLANT
DESIGNS"
MARCH 28, 1994

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# POLICY ISSUE
(Notation Vote)

March 28, 1994 SECY-94-084

**FOR:** The Commissioners

**FROM:** James M. Taylor
Executive Director of Operations

**SUBJECT:** POLICY AND TECHNICAL ISSUES ASSOCIATED WITH THE REGULATORY TREATMENT OF NON-SAFETY SYSTEMS IN PASSIVE PLANT DESIGNS

**PURPOSE:**

To present the Commission with recommended positions pertaining to policy and technical issues affecting passive advanced light water reactor (ALWR) designs and to request that the Commission approve the underlined staff positions presented in this paper.

**SUMMARY:**

In the enclosure, the U. S. Nuclear Regulatory Commission (NRC) staff discusses eight technical and policy issues pertaining to the regulatory treatment of non-safety systems (RTNSS) for passive ALWRs. The staff previously identified these issues in the draft Commission papers, "Issues Pertaining to Evolutionary and Passive Light-Water Reactors and Their Relationship to Current Regulatory Requirements," February 20, 1992, and "Design Certification Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light-Water Reactor Designs," June 25, 1992; and in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," April 2, 1993. After extensive dialogue with the Electric Power Research Institute (EPRI), the vendors, and the Advisory Committee on Reactor Safeguards (ACRS), the staff proposed its position on these technical and policy issues in a draft Commission paper issued September 7, 1993. Subsequently, comments were received from EPRI and from

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NOTE: TO BE MADE PUBLICLY AVAILABLE IN 3 WORKING DAYS FROM THE DATE OF THIS PAPER

Westinghouse. The staff briefed the ACRS in August and November on these issues. The ACRS comments on the eight policy and technical issues associated with RTNSS were provided to the Chairman in a letter dated November 10, 1993. After considering industry, vendor, and ACRS comments, the staff has reached a final position on the RTNSS issues. The staff has underlined the positions for which it is requesting the Commission's approval.

**BACKGROUND:**

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 the staff is substantially supplementing or revising interpretive guidance applied to currently-licensed light water reactors (LWRs). The staff considers these to be policy issues fundamental to agency decisions on the acceptability of ALWR designs.

As described in the summary above, the eight technical and policy issues associated with RTNSS have been previously identified to the Commission.

In SECY-93-087, the staff indicated that it would be discussing control room habitability in a Commission paper on the subject of source term. Although control room habitability is linked to passive plant policy and source term issues, the staff believes that it was more appropriate to discuss control room habitability as a passive plant issue; hence, recommendations on control room habitability are presented in this paper.

In SECY-93-087, the staff also provided the Commission its interim position on the reliability assurance program (RAP) applicable to design certification. The staff stated that the final position on RAP would be included in a future Commission paper on the regulatory treatment of non-safety systems. This paper provides the staff's position on RAP for both the evolutionary and passive ALWRs.

**DISCUSSION:**

The regulatory treatment of non-safety-related systems in advanced reactor passive designs will have wide-ranging effects on both the design and licensing of the AP600 and the simplified boiling water reactor (SBWR). Unlike the current generation of LWRs or the evolutionary ALWRs, the passive ALWR designs make extensive use of safety systems that rely on the driving forces of buoyancy, gravity, and stored energy sources. These passive systems supply safety-injection water, perform core and containment cooling, and perform other functions. These passive safety systems contain no pumps and include valves that are operated by either air pressure or dc electric power from batteries, or use check valves actuated by the pressure differential across the valve. In addition to the active systems used during normal plant operations, 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 are the first line of defense to reduce challenges to the passive systems in the event of transients or plant upsets.

The licensing design-basis analyses proposed by the industry for the passive designs rely solely on the passive safety systems to demonstrate compliance with the acceptance criteria of various design-basis transients and accidents. 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 AP600 and SBWR submittals.

The enclosure discusses the staff's position, the current regulatory requirement or interpretations, and comments received from industry and vendors regarding eight technical and policy issues pertaining to the RTNSS for passive ALWR designs, including RAP. The RAP also applies to evolutionary ALWR designs. The staff has included a discussion of the basis for its position on each issue. The staff underscored the positions for which it is requesting the Commission's approval.

The staff development of the staff positions was based on the following:

(1) review of the available information on passive ALWR designs;

(2) consideration of insights from the available results of the probabilistic risk assessments (PRAs) of LWRs and ALWRs;

(3) completion of the safety evaluation report for the EPRI utility requirements document (URD) for passive ALWR designs;

(4) consideration of EPRI and industry comments on these issues which were raised during a meeting between NRC staff and the ALWR Steering Committee on January 22, 1993, in Palo Alto, California, and in meetings between NRC staff and EPRI representatives on April 15, and May 20, 1993, in Rockville, Maryland;

(5) review of EPRI's letters of February 23, and May 13 and 26, 1993, which detailed a proposed process for the RTNSS in passive plant designs; and

(6) consideration of EPRI, ACRS, and industry comments on a draft version of this paper which was forwarded to the Commission on September 7, 1993.

The staff concludes that the positions discussed in the enclosure are fundamental to the Agency's decisions on the acceptability of the passive LWR designs (and on the RAP for evolutionary plant 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.

The staff proposes to make this paper and its enclosure available to the public no sooner than 3 work days after this paper is forwarded to the Commission.

**CONCLUSIONS:**

The staff requests that the Commission approve the recommended positions for issues pertaining to the regulatory treatment of non-safety systems in passive advanced light water designs. This will enable the staff to proceed more effectively with its review of Westinghouse's AP600 and GE Nuclear Energy's simplified boiling water reactor ALWR designs and, in the case of RAP, resolve the evolutionary ALWR design reviews.

**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 vendors and EPRI regarding these Commission positions should state this fact.

The ACRS was briefed on August 5, and November 4, 1993. The ACRS provided its comments on the draft Commission paper issued September 7, 1993, in a letter to the Chairman dated November 10, 1993. In a letter dated February 2, 1994, the staff responded to the ACRS comments. Those responses are reflected in the positions contained in the enclosure to this paper.

Additional comments on RAP were provided by the ACRS in its letter dated February 17, 1994, and the staff will be responding separately. The staff's views on the ACRS concerns are reflected in the enclosure to this paper. We continue to believe that RAP provides a useful process to allow probabilistic and deterministic risk insights to be considered during the design and operation of ALWRs and is not inconsistent with provisions of the Maintenance Rule. The staff also agrees with the ACRS in the matter of infeasibility of demonstrating plant-specific structure, system, and component reliability.

**RECOMMENDATIONS:**

The staff recommends that the Commission

(1) Approve the positions underlined in the enclosure.

(2) Note that the staff will make the enclosure available to the public no sooner than 3 work days after this paper is forwarded to the Commission. The staff will indicate that the proposed resolutions are being considered by the Commission, and therefore, are not final positions.

James M. Taylor
Executive Director
for Operations

Enclosure:
Policy Issues Analysis
and Recommendations
for Passive Plants

Commissioners' comments or consent should be provided directly to the Office of the Secretary by COB Tuesday, April 12, 1994.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT Tuesday, April 5, 1994, 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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## POLICY ISSUES ANALYSIS AND RECOMMENDATIONS FOR PASSIVE PLANTS

### A. Regulatory Treatment of Non-safety Systems

Unlike the current generation of light water reactors or the evolutionary advanced light water reactors (ALWRs), the passive ALWR designs use passive safety systems that rely exclusively on natural forces, such as density differences, gravity, and stored energy to supply safety injection water and provide core and containment cooling. These passive systems do not include pumps. All valves in these passive systems either require only dc electric power by means of batteries, are operated by air pressure, or are check valves operating by means of pressure differential across the valve. These passive systems do not receive safety-related ac electric power. The designers designate all the active systems as non-safety systems except for limited portions of the systems that provide safety-related isolation functions such as containment isolation.

As the passive ALWR designs rely on the passive safety systems to perform design-basis safety functions of reactor coolant makeup and decay heat removal, different portions of the passive systems also provide certain defense-in-depth backup to primary passive features. For example, while the passive decay heat removal heat exchanger is the primary safety-related heat removal feature in a transient, the automatic reactor depressurization system together with the passive safety injection features provide a safety-related defense-in-depth backup.

The passive ALWR designs also include active systems that provide defense-in-depth capabilities for reactor coolant makeup and decay heat removal. These active systems are the first line of defense to reduce challenges to the passive systems in the event of transients or plant upsets. As stated above, all active systems in passive plants are designated as non-safety systems. In addition, one of the principal design requirements of EPRI's ALWR utility requirements document (URD) is that passive systems should be able to perform their safety functions, independent of operator action or offsite support, for 72 hours after an initiating event. After 72 hours, non-safety, or active systems may be required to replenish the passive systems or perform core and containment heat removal duties directly. As specified in the URD, these active systems which may be needed to provide defense-in-depth capabilities include (1) the chemical and volume control system and control rod drive system, which provide reactor coolant makeup for the passive pressurized water reactor (PWR) and boiling water reactor (BWR), 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 to support these functions, including non-safety standby diesel generators. The ALWR URD also requires that the plant designer specifically define the active systems relied on for defense-in-depth for a standard design as necessary to meet passive ALWR plant safety and investment goals. These active systems may include additional systems beyond those discussed above. The passive ALWR designs also include other active systems, which are designated as non-safety, (such

as the heating, ventilation, and air conditioning (HVAC) system) that remove heat from the instrumentation and control (I&C) cabinet rooms and the main control room and prevent the excessive accumulation of radioactive materials in the control room to limit challenges to the passive safety capabilities for these functions.

In existing plants (and in evolutionary ALWR designs), the NRC has treated many of these active systems as safety-related systems. As stated earlier, active systems are not classified as safety-related in passive ALWR designs, and credit is not taken for these active systems in the Chapter 15 licensing design basis accident (DBA) analyses. In SECY-90-406, "Quarterly Report on Emerging Technical Concerns," December 17, 1990, the staff listed the role of these active systems in the passive design as an emerging technical issue. In SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs," April 2, 1993, the staff discussed the issue of regulatory treatment of active non-safety systems (the "RTNSS Issue") and stated that it would propose a resolution of this issue in a separate Commission paper.

Because of limited operational experience and the low-driving force of the passive safety systems, the designers have not verified all aspects of the passive features and the overall capabilities of reactor coolant makeup and core and containment heat removal. The passive systems involve inherent phenomenological uncertainties such as those associated with the performance of check valves operating under natural circulation or gravity injection with low differential pressures that may not create sufficient force to fully open a stuck check valve, unlike the emergency core cooling systems in current operating plants in which pressure developed by pumps can overcome stuck valves. The staff expects these uncertainties to be reduced through carefully planned and implemented components performance tests, and separate effects and integral system tests, and/or prototype tests over a sufficient range of transient and accident conditions per 10 CFR 52.47(b)(2)(i)(B), combined with realistic analyses of the performance of passive systems and components for these ALWRs.

The residual uncertainties associated with passive safety system performance increase the importance of active systems in providing defense-in-depth functions to the passive systems. The NRC staff and EPRI have developed a process for maintaining appropriate regulatory oversight of these active systems in the passive ALWR designs. The staff will not require that these active systems meet all the safety-related criteria, but will expect a high level of confidence that active systems which have a significant safety role are available when challenged.

The ALWR URD specifies requirements concerning design and performance of active systems and equipment that perform non-safety, defense-in-depth functions. These requirements include radiation shielding to permit access after an accident, redundancy for the more probable single active failures, availability of non-safety-related electric power, and protection against more probable hazards. The requirements also address realistic safety margin basis analysis and testing to demonstrate the systems' capability to satisfy their

non-safety defense-in-depth functions. EPRI has proposed that the ALWR URD will not include specific requirements for the quantitative reliability of these systems.

The exclusive reliance on passive systems in meeting current licensing criteria is a departure from current design philosophy and licensing practice and must be evaluated. Therefore, the staff will need new guidance for reviewing the AP600 and SBWR submittals and in developing regulatory treatment of non-safety systems (RTNSS).

The staff met with representatives of the ALWR Program on several occasions to determine the steps needed to resolve the issue of RTNSS in passive plants, and define the scope of requirements and acceptance criteria to ensure that they have adequate capability and availability, when required. In a meeting between NRC and the ALWR Utility Steering Committee on January 22, 1993, the participants agreed to an overall process for determining the regulatory treatment of non-safety systems, and determining the importance of passive systems and components for meeting NRC safety objectives. This agreement included the following key elements:

1. EPRI has proposed that the passive ALWR URD will describe the process to be used by the designer for specifying the reliability/availability (R/A) missions of risk-significant structures, systems, and components (SSCs) needed to meet regulatory requirements and to allow comparison with NRC safety goals. An R/A mission is the set of requirements related to performance, reliability, and availability for an SSC function that adequately ensure its task, as defined by the focused PRA or deterministic analysis, is accomplished. The focused PRA is described in Section II.3, below.

2. The designer will apply the process to the design to establish R/A missions for the risk-significant SSC.

3. If active systems are determined to be risk significant, NRC will review these R/A missions to determine if they are adequate and if the operational reliability assurance program (O-RAP) or simple technical specifications and limiting conditions for operation are adequate to give reasonable assurance that the missions can be met during operation.

4. If active systems are relied on to meet the R/A missions, the designer will impose design requirements commensurate with risk significance on those elements involved.

5. NRC will not include any R/A missions in the design certification rule. Instead, NRC would include deterministic requirements on both safety and non-safety design features in the design certification rule.

To address these key elements, the staff and representatives of the ALWR Program later began preparing an appropriate process that the plant designers can use to address the RTNSS issue. In a letter of February 23, 1993, the ALWR Program submitted a proposed process for determining the appropriate regulatory treatment for active systems for passive ALWRs. In a meeting on

May 20, 1993, the staff and representatives of the ALWR Program agreed to a final process for resolving the RTNSS issue. In a letter of May 26, 1993, EPRI described the steps in this process for determining risk-significant non-safety features based on a Level 3 probabilistic risk assessment (PRA). The process involves constructing a "focused PRA" to determine the importance of various active systems in ensuring that the Commission's safety goal objectives are met. Risk-significant SSCs, their R/A missions, and regulatory oversight can then be determined. The steps of this RTNSS process described by EPRI in their May 26, 1993, submittal are as follows:

I. Scope and Criteria

The RTNSS basis applies broadly to those non-safety SSCs that perform risk-significant functions, and therefore, are candidates for regulatory oversight. The plant designer will apply the following criteria, proposed by EPRI in their May 26, 1993, submittal, to determine these SSC functions:

A. SSC functions relied upon to meet beyond design basis deterministic NRC performance requirements such as 10 CFR 50.62 for anticipated transient without scram (ATWS) mitigation and 10 CFR 50.63 for station blackout.

B. SSC functions relied upon to resolve long-term safety (beyond 72 hours) and to address seismic events.

C. SSC functions relied upon under power-operating and shutdown conditions to meet the Commission's safety goal guidelines of a core damage frequency of less than 1.0E-4 each reactor year and large release frequency of less than 1.0E-6 each reactor year.

D. SSC functions needed to meet the containment performance goal (SECY-93-087, Issue I.J), including containment bypass (SECY-93-087, Issue II.G), during severe accidents.

E. SSC functions relied upon to prevent significant adverse systems interactions.

The staff finds the proposed scope and criteria to be acceptable. It should be noted that the large release frequency of less than 1.0E-6 each reactor year specified in Item C, above, as one of the screening criteria was an agreement reached between the NRC and the ALWR Steering Committee and was proposed in the May 26, 1993, EPRI submittal. Subsequently, the Commission has decided to terminate the development of the definition of large release. Therefore, the staff will work with the ALWR vendors to assess the need for any alternative criterion. A conditional containment failure probability of 0.1 was previously approved by the Commission as a complement to the deterministic containment performance goal.

II. Specific Steps in the RTNSS Process for Each Design

1. Comprehensive Baseline PRA

The designer will construct comprehensive Level 3 PRAs (baseline PRAs) in accordance with the ALWR URD. These comprehensive baseline PRAs must include all appropriate internal and external events for both power and shutdown operations. Seismic events will be evaluated by a margins approach. Adequate treatment of uncertainties, long-term safety operation, and containment performance should be included. Containment performance should be addressed with considerations for sensitivities and uncertainties in accident progression and inclusion of severe accident phenomena, including explicit treatment of containment bypass. Mean values must be used to determine the availability of passive systems and the frequencies of core damage and large releases. Appropriate uncertainty and sensitivity analyses should be used to estimate the magnitude of potential variations in these parameters and to identify significant contributors to these variations. Results of an adverse systems interaction study will also be considered in the PRA.

2. Search for Adverse Systems Interactions

The designers must systematically evaluate adverse interactions between the active and passive systems. The results of this analysis should be used for design improvements to minimize adverse systems interaction, and be considered in making PRA models.

3. Focused PRA

The focused PRA includes the passive systems and only those active systems necessary to meet the safety goal guidelines proposed by EPRI in scope Criteria I.C. The designers should consider the following in constructing focused PRAs to determine the R/A missions of non-safety SSCs which are risk significant.

First, the scope of initiating events and their frequencies are maintained in the focused PRA as in the baseline PRA. As a result, non-safety SSCs used to prevent the occurrence of initiating events will be subject to regulatory oversight applied commensurate with their R/A missions for prevention, as discussed in Steps 4 and 5, below.

Second, following an initiating event, the comprehensive Level 3 focused PRA event tree logic will not include the effect of non-safety SSCs. As a minimum, these event trees will not include the defense-in-depth functions and their support such as ac power to determine if the passive safety systems, when challenged, can provide sufficient capability without non-safety backup to meet the NRC safety goal guidelines for a core damage frequency of 1.0E-4 each year and a large release frequency of 1.0E-6 each year. The designer should evaluate the containment performance, including bypass, during a severe accident. Non-safety SSCs which remain in the focused PRA model are subject to regulatory oversight based on their risk significance in Steps 4 and 5.

4. Selection of Important Non-safety Systems

The designers will determine any combinations of non-safety SSCs that are necessary to meet NRC regulations, safety goal guidelines, and the containment performance goal objectives. The designers will determine these combinations for both scope Criteria A and E where NRC regulations are the bases for consideration and scope Criteria C and D where PRA methods are the bases for consideration. To address the long-term safety issue in scope Criterion B, the designer will use PRA insights, sensitivity studies, and deterministic methods to establish the ability of the design to maintain core cooling and containment integrity beyond 72 hours. Non-safety SSC functions required to meet beyond design basis requirements (Criterion A), to resolve the long-term safety and seismic issues (Criterion B), and to prevent significant adverse interactions (Criterion E) are subject to regulatory oversight as discussed in Step 6, below.

EPRI has proposed that the designers will take the following steps in using the focused PRA to determine the non-safety SSCs important to risk:

a. Determine those non-safety SSCs needed to maintain initiating event frequencies at the comprehensive baseline PRA levels.

b. Add the necessary success paths with non-safety systems and functions in the "focused PRA" to meet the safety goal guidelines, containment performance goal objectives, and NRC regulations. Choose the systems by considering the factors for optimizing the design effect and benefit of particular systems. Perform PRA importance studies to assist in determining the importance of these SSCs. Recognize that the staff could require regulatory oversight for all non-safety SSCs in the focused PRA model needed to meet NRC requirements, the safety goal guidelines, and containment performance goals.

5. Non-safety System Reliability/Availability Missions

The designers will determine and document from the focused PRA the functional R/A missions of active systems needed to meet the safety goal guidelines, containment performance goals, and other NRC performance requirements as described in Step 4. Repeat Steps 4, 5 and 6 to ensure that the best active systems and their R/A missions are selected.

As part of this step, the designer should establish graded safety classifications and graded requirements for I&C systems based on the importance to safety of their functional R/A missions. In SECY-91-292, the staff discussed the need for such classifications and requirements for I&C systems important to safety.

6. Regulatory Oversight Evaluation

Upon completing Steps 1-5, above, the designers will conduct activities such as:

a. Reviewing the standard safety analysis report (SSAR) and the PRA, and audit plant performance calculations to determine that the design of these risk-significant non-safety SSCs satisfies the performance capabilities and R/A missions.

b. Reviewing the SSAR to determine that it includes the proper design information for the reliability assurance program, including the design information for implementing the maintenance rule and operational reliability assurance program.

c. Reviewing the SSAR to determine that it includes proper short-term availability control mechanisms, if required for safety and determined by risk significance such as simple technical specifications.

After the designer has completed these or related activities, the staff will apply appropriate regulatory oversight.

7. NRC/Vendor Interaction

Early in the reviews, the staff and the designers will discuss the appropriateness of the focused PRA models and reliability values, R/A missions, and level of regulatory oversight for various active systems.

This process which EPRI has proposed for RTNSS was developed after several meetings with the NRC staff. The staff endorses the process described in this paper and finds it to be an acceptable method for handling the RTNSS issue.

As a part of NRC/EPRI agreement, EPRI will properly incorporate this RTNSS process in the ALWR URD for the passive plant designer to address the RTNSS issue. However, the risk significance of active systems cannot be determined until the design-specific baseline and focused PRA evaluation are completed because the design requirements of active systems depend on the R/A missions of the risk-significant active systems, which the plant designer will determine using the RTNSS process and the design-specific focused PRA. The staff cannot complete portions of its review for the performance goals of both passive and active systems, technical specification requirements, and the operational reliability assurance program before the designers submit the focused evaluation described above and before the PRA review is nearly completed to determine the R/A missions. These actions must be completed in a timely manner to ensure the designers and prospective owner/operators understand the results of these reviews and their implications on operational regulatory requirements in time to accommodate the requirements or explore alternative measures.

The designer must integrate into the design process the process for resolving the RTNSS issue. In particular, the designer should use the results from identifying the risk-significant important systems and their R/A missions and comparisons with the safety goal objectives, and report this information in the PRA. By including this information in the review of the PRA and related discussions with the designer, the staff will determine the regulatory oversight on the non-safety SSCs in the most efficient and timely way.

This RTNSS process is a comprehensive approach for resolving the RTNSS issue and other relevant issues evaluated in the process. In determining the R/A missions and the proper regulatory oversight of the risk-significant active systems during this evaluation, the staff will properly address these issues, which include the stable safe shutdown requirements and related passive system design basis of 72-hour capability, station blackout, electrical distribution, and control room habitability and inservice testing of pumps and valves.

The staff recommends that the Commission approve the proposed process as an acceptable method for resolving the regulatory treatment of non-safety systems in the passive ALWR designs.

**B. Definition of Passive Failure**

A single failure is defined in Appendix A to 10 CFR Part 50 as an occurrence which results in the loss of a component's capability to perform its intended safety functions. The deterministic single failure criterion is a simple, effective method to determine the redundancy of systems and components needed to ensure adequate reliability of safety functions. General experience indicates that even components and equipment that are made to high standards of quality may sometimes fail to function in a way and at a time that can be random and unpredictable.

The NRC regulations include the single failure criterion in the general design criteria (GDC) in Appendix A to 10 CFR Part 50, which require the design of certain systems important to safety to be capable of performing their defined safety functions or mission assuming the failure of any single component within the system or its supporting systems. For example, GDC 21, 34, and 35, respectively, require sufficient redundancy and independence to be designed into the protection, residual heat removal, and emergency core cooling systems such that no single failure results in the loss of these system safety functions.

In SECY-77-439, "Single Failure Criterion," the staff described how it is using the single failure criterion in reviewing reactor safety. Though the NRC established the single failure criterion without assessing the probabilities of component or system failure, it is not assumed that any conceivable failure could occur in applying the criterion. In general, only those systems or components judged to have a credible chance of failure are assumed to fail in applying the single failure criterion.

In SECY-77-439, the staff discussed the distinction between active and passive failures of a system or component. An active failure in a fluid system is (1) the failure of a component which relies on mechanical movement 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, the spurious opening or closing of a motor- or air-operated valve, or the failure of a pump to start or stop on demand. Such failures can be induced by operator error. A passive failure in a fluid system is a breach in the fluid pressure boundary or a mechanical failure which adversely affects a flow path. Examples include the failure of a check valve to move to its correct position when required and 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 defining a single failure in Appendix A to 10 CFR Part 50, the NRC stated that fluid and electric systems are considered to be designed against an assumed single failure if the system maintains its ability to perform its safety functions in the event of either (1) a single failure of any active component (assuming passive components function properly) or (2) a single failure of a passive component (assuming active components function properly). The NRC further noted that single failures of passive components in electric systems should be assumed in designing against a single failure. Thus, no distinction is made between failures of active and passive components for electric systems, and all such failures must be considered in applying the single failure criterion. Appendix A also states 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 being developed.

In SECY-77-439, the staff stated the following:

> on the basis of the licensing review experience accumulated in the period since 1969, it has been judged in most instances that 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 the application of single failure criterion to assure safety of a nuclear power plant.

In keeping with the defense-in-depth approach, the staff does consider the effects of certain passive failures (e.g., check valve failure, medium- or high-energy pipe failure, and valve stem or bonnet failure) as potential accident initiators. In licensing reviews, however, only on a long-term basis does the staff consider passive failures in fluid systems as potential accident initiators in addition to initiating events. For example, Section 6.3 of the Standard Review Plan (SRP) requires consideration of passive failures in the emergency core cooling system during the recirculation cooling mode following emergency cooling injection, but does not define such a failure. The staff finds no reason to alter this regulatory practice for the passive ALWR designs, except for check valves as discussed below.

The failure of a check valve to move to its desired position is not clearly defined as an active or passive failure. American National Standards Institute (ANSI)/American Nuclear Society (ANS)-58.9 cites the failure of a check valve to move to its correct position as an active failure. In SECY-77-439, the staff stated that the failure of a check valve to move to its correct position when required was a passive failure. The staff normally treats check valves, except for those in containment isolation systems, as passive devices. In an International Atomic Energy Agency (IAEA) paper, "Application of the Single Failure Criterion - A Safety Practice," (Safety Series 50-P-1) the authors stated that in some member States a failure of a simple swing type check valve to open need not be considered as a single failure, whereas in

other member States self-operating components such as check valves are considered to be active components if the state of the component is changed during the given event sequence after an initiating event. The authors of the IAEA paper determined that, with the test intervals of check valves of about one year, the probabilities of failure of check valves to open or close are closer to the failure probabilities of active components (3E-6 to 3E-5 per hour) than to those of passive components (1E-9 to 1E-8 per hour). The authors stated that a conservative approach is to assume a check valve failure in the single failure analysis.

For current plants, the NRC staff normally treats check valves, except for those in containment isolation systems, as passive devices during transients or design-basis accidents. Therefore, the staff would not consider the failure of a check valve to be a single active failure. Recognizing the unique features of the passive safety system designs having low-driving force, the staff examined current regulatory practice to determine how it will apply to check valve failures for the passive plant designs. These safety-related check valves in the passive designs will operate under different conditions (low flow and pressure without pump discharge pressure to open valves) than current generation reactors and evolutionary designs. Check valves have high safety significance in the operation of the passive safety systems, and operating experience of check valves suggests that they may have a lower reliability than originally anticipated.

EPRI stated that the ALWR program endorses ANSI/ANS 58.9-1981 considering check valves to be active components when they are required to change state to perform their safety function. Failures of these components are considered to be active failures that occur coincident with event initiators. The ANS standard allows exemptions where the proper function of a component can be demonstrated despite any credible condition, and it requires documentation of the exemptions in the single failure analysis. EPRI further stated that the ALWR reliability program will include a thorough review of check valve applications in the passive safety systems. This will include determining the particular check valves which play a key role in ensuring core damage frequency requirements are met, reviewing whether available check valve reliability data is applicable and sufficient for passive plant safety systems, and determining appropriate measures for assuring that check valves will operate reliably throughout the plant operating life. EPRI stated that its intent is to rigorously evaluate these valves to establish the best technical solution rather than simply relying on single failure to ensure safety. In a position paper, "NRC Policy Issue Analysis and Recommendation," submitted with a May 5, 1992, letter, EPRI contended that check valves when appropriately designed for the application will be extremely reliable. EPRI also contended that the URD requirements, ALWR safety goals, and the iterative use of PRA in the design process ensure that the unavailability of check valves will be sufficiently low and independent of the initiating failure that check valves need not be assumed to fail. EPRI recommended that check valve failures not be redefined as active failures for the passive safety systems. In its letter of December 10, 1992, EPRI also stated that this industry position is consistent with ANS 58.9, which appears to be inconsistent with

EPRI's earlier endorsement of ANSI/ANS 58.9-1981 that considers check valves as active components if they must change state to perform their safety function.

The staff proposes that, except for those check valves whose proper functions can be demonstrated and documented, check valves in the passive safety system designs be subject to single active failure consideration. In determining an exemption to single failure consideration for a particular check valve application, the plant designer shall perform a comprehensive evaluation of check valve test data or operational data for the similar check valve designs in similar applications and operating environments to demonstrate that the reliability of the particular check valve application is such that the probability of failure is comparable to those of passive components. A failure probability on the order of 1E-4 per year or less would be low enough to be considered as a passive failure. An example of possible exemption is the accumulator check valves installed in applications identical to those for currently licensed plants where the accumulator pressure will eventually create a large pressure differential to force open the valves as the reactor coolant system (RCS) pressure falls.

Redefining check valves as active components, subject to consideration for single active failures would cause these valves to be evaluated in a more stringent manner than that used in previous licensing reviews.

The staff recommends that the Commission approve the staff's proposal to maintain the current licensing practice for passive component failures on the passive ALWR designs, and to redefine check valves, except for those whose proper function can be demonstrated and documented, in the passive safety systems as active components subject to single failure consideration.

**C. Safe Shutdown Requirements**

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

The NRC regulations have several definitions for safe shutdown. For example, in 10 CFR 50.2, the NRC regulations define "safe shutdown (non-design basis accident)" for station blackout as bringing the plant to those shutdown conditions specified in plant technical specifications as hot standby or hot shutdown, as appropriate (plants have the option of maintaining the RCS at normal operating temperatures or at reduced temperatures). Appendix R to 10 CFR Part 50 states that the phrase "safe shutdown" is used throughout the appendix as applying to both hot and cold shutdown. The regulation does not define safe shutdown of the plant after normal operation or a design basis

accident, nor does it define what constitutes a safe shutdown state. In implementing the GDC 34 requirements, the staff specified in Regulatory Guide (RG) 1.139 "Guidance for Residual Heat Removal," and Branch Technical Position (BTP) RSB 5-1 the conditions for cold shutdown (93.3 °C (200 °F) for a PWR and 100 °C (212 °F) for a BWR) using only safety-grade systems within 36 hours. In the regulatory guide, the staff 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 are limited by the inherent ability of the passive heat removal processes because they use passive heat removal systems for decay heat removal. These designs cannot reduce the temperature of the reactor coolant system below the boiling point of water for the heat to be transferred to the water pool where heat exchangers are submerged, that is, the in-containment refueling water storage tank of the AP600 or the isolation condenser of the simplified boiling water reactor (SBWR). Even though active shutdown cooling systems are available to bring the reactor to cold shutdown or refueling conditions, these active RHR systems are not safety-grade and do not comply with the guidance of RG 1.139 or BTP RSB 5-1.

EPRI defined a safe stable shutdown condition as 215.6 °C (420 °F) and stated that passive safety systems need not be capable of achieving cold shutdown. EPRI based this contention on the belief that the passive decay heat removal systems have an inherently high long-term reliability. EPRI contended that the passive ALWR designs meet the GDC 34 requirements because they use a redundant safety-grade passive system that can operate at full RCS pressure and place the reactor in the long-term cooling modes immediately after shutdown, and because conditions maintained by the systems are safe and fully consistent with the GDC 34 requirement to maintain fuel and reactor coolant pressure boundary within acceptable limits.

In evaluating the EPRI position on safe shutdown, the staff considered the conditions that constitute a safe shutdown state and assessed the acceptability of EPRI's proposed approach for meeting GDC 34. In RG 1.139 and BTP 5-1, the staff position that an RHR system be able to bring the plant to cold shutdown conditions was to enable the licensee to perform inspection and repair at the plant. The staff believes that other plant conditions may constitute a safe shutdown state as long as reactor subcriticality, decay heat removal, and radioactive materials containment are properly maintained for the long term.

The URD for passive designs specifies performance requirements for the passive decay heat removal systems to have sufficient capacity to reduce reactor

coolant temperature to 215.6 °C (420 °F) within 36 hours of reactor shutdown. To ensure the means are available to remove decay heat in accordance with GDC 34, the URD also specifies that, upon a single failure, safety-grade decay heat removal from the reactor coolant system shall be possible from full RCS operating pressures and temperatures to a safe stable condition for all plant conditions. EPRI also required that the operation of the plant in the long-term cooling mode be automatic, eliminating the need for operator actions to cool down the plant. The operation of the passive RHR system does not require ac power, pump, or valve operation (except for initial operation for alignment of the system), or support systems (such as component cooling water or service water), and is stable and self-contained, requiring no makeup water for a period of at least 3 days following reactor shutdown. Therefore, the licensee could maintain a safe stable condition with the safety-grade passive RHR system.

After the passive RHR system or main steam system effected the initial shutdown, a non-safety-grade reactor shutdown cooling system will be available to bring the plant to cold shutdown conditions for inspection and repair. EPRI stated that

> these non-safety systems are required to be highly reliable. . .and there is no single failure of these systems or their support systems which would result in inability to terminate use of the passive safety grade system and achieve cold shutdown if desired.

The staff believes that the passive RHR systems offer potential advantages over current active systems, and can maintain the plant in conditions that are fully consistent with the requirement of GDC 34 to maintain the fuel and reactor coolant pressure boundary within acceptable limits, and therefore, contain radioactive materials which may be present. The passive safety injection system and the associated depressurization system can also protect against the loss of reactor coolant inventory during long-term passive RHR operation. These passive system capabilities can be demonstrated by appropriate evaluations during detailed design analyses, including

1. A safety analysis to demonstrate that the passive systems can bring the plant to a safe stable condition and maintain this condition, that no transients will result in the SAFDLs and pressure boundary design limit being violated, and that no high-energy piping failure being initiated from this condition will result in violation of 10 CFR 50.46 criteria.

2. A probabilistic reliability analysis, including events initiated from the safe shutdown conditions, to ensure conformance with the safety goal guidelines. The PRA would also determine the R/A missions of risk-significant systems and components as a part of the effort for regulatory treatment of non-safety systems.

The staff is concerned that, with the passive system design basis of 72-hour capability, the passive RHR system water pool, without refill, will have water capacity to permit only 72 hours of operation after a scram. A long-term safe stable condition, however, can be maintained if a reliable non-safety support

system or equipment is available to replenish the water pool to sustain long-term operation of the passive RHR system after 72 hours. The passive URD requires that non-safety equipment necessary for plant recovery after the assumed 72-hours accident duration be designed for the expected environment, and that only simple, unambiguous operator actions and easily accomplished offsite assistance be necessary after 72 hours to prevent fuel damage. The staff recommended in Section A of this paper that the Commission approve an acceptable process for resolving the RTNSS issue. With an acceptable resolution of the RTNSS issue, the staff expects that non-safety support systems and equipment and active decay heat removal systems will be evaluated for their risk significance and will meet appropriate design and reliability criteria to provide backup capability to passive systems beyond 72 hours. This will ensure proper operation of the passive RHR system to maintain a safe stable condition over the long term, as well as reliable non-safety systems that will be necessary to bring the plant to cold shutdown conditions.

The staff concludes that cold shutdown is not the only safe stable shutdown condition which can maintain the fuel and reactor coolant boundary within acceptable limits, and that the EPRI proposed 215.6 °C (420 °F) as a safe stable shutdown condition is acceptable on the basis of acceptable passive safety system performance and acceptable resolution of the regulatory treatment of non-safety systems.

The staff recommends that the Commission approve the EPRI's proposed 215.6 °C (420 °F) or below, rather than the cold shutdown condition required by RG 1.139, as a safe stable condition, which the passive decay heat removal systems must be capable of achieving and maintaining following non-LOCA events. This recommendation is predicated on an acceptable passive safety system performance and an acceptable resolution of the issue of regulatory treatment of non-safety systems.

**D. Control Room Habitability**

GDC 19 of Appendix A to 10 CFR Part 50 states that (1) a control room should be provided from which actions can be taken to operate the nuclear power plant safely under normal conditions and to maintain it in a safe condition under accident conditions including a loss-of-coolant accident and (2) adequate radiation protection should be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. In current plants, safety-grade, filtered control room HVAC systems with charcoal adsorbers are used to ensure that radiation doses to operators will be maintained within the GDC 19 criteria in the event of an accident.

In SRP Section 6.4, "Control Room Habitability Systems," the staff defined the acceptable operator dose criteria in terms of specific whole-body and critical organ doses (5 rem to the whole body and 30 rem each to the thyroid and skin).

Originally, EPRI proposed the exposure limit for control room operators of 5 rem whole body, 75 rem beta skin dose, and 300 rem thyroid dose. EPRI

stated that each operator would be provided with individual breathing apparatus and protective clothing, if required, to meet regulatory limits. The staff determined that EPRI did not adequately justify its requirements for the thyroid and beta skin doses. The staff informed EPRI that the long-term use of breathing apparatus during design-basis accidents has never been allowed. More importantly, the long-term use of breathing apparatus is likely to degrade control room operator performance during and after an accident.

EPRI stated that the control room would be designed to be maintained during a 72-hour period as the primary location from which personnel can safely operate in the event of an accident. The staff's position is that the required duration for certain accident sequences may 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," which has typically been assumed to be 30 days. Consequently, the staff concluded that analyses of control room habitability should consider the duration of the accident which may extend beyond the EPRI-proposed 72-hour period as the design basis.

In its letter of May 5, 1992, EPRI proposed an alternative in which a safety-grade pressurization system could be recharged remotely after 72 hours. The URD for passive plants requires (1) a passive, safety-grade control room pressurization system which would use bottled air to keep operator doses within the limits of GDC 19 and SRP 6.4, Revision 2 of the SRP for the first 72 hours of the event, and (2) safety-grade connections for the pressurization system to allow the use of offsite, portable air supplies if needed after 72 hours to minimize operator doses. The staff agrees with the concept of a safety-grade pressurization system and EPRI's commitment to limit the operator doses to those specified in GDC 19 and SRP 6.4, Revision 2. The staff will evaluate the feasibility and the capability of the proposed pressurization systems on a vendor-specific basis.

In its letter of August 17, 1992, the Advisory Committee on Reactor Safeguards (ACRS) stated that the members had discussed control room habitability with EPRI and the staff during a June 4 and 5, 1992, meeting. At that meeting, the staff told the ACRS that it was evaluating the EPRI proposal for the safety-grade pressurization system. ACRS stated that it had several comments about the design features of the passive control room pressurization system proposed by EPRI. The ACRS stated that the staff should consider these comments when performing its evaluation and that the ACRS may make additional recommendations after the staff has completed its evaluation. In an October 29, 1992, reply to the ACRS, the staff stated that it had not completed its review of the control room habitability issue and would consider the ACRS comments during its review of the EPRI Requirements Document.

The staff reviewed the EPRI proposal for a safety-grade pressurization system and determined the following:

- The present licensing of nuclear power plants does not require the licensee to have engineered safety feature (ESF) ventilation systems unless the licensee cannot meet the dose criteria associated with the design basis accidents (DBAs) or other safety criteria. If the licensee

cannot meet these criteria, it must ensure that an ESF system or some other safety-grade system is available to mitigate the consequences of a DBA.

- The use of a pressurization system, such as a bottled air system, may not preclude the need for other safety-grade ventilation equipment within the control room. For example, such safety-grade equipment could be required to maintain cooling to the electrical instruments in the control room.

- At least once each refueling cycle, the licensee must demonstrate the adequacy of such a system to pressurize the control room for a 72-hour period and maintain all the other conditions, including temperature, within the acceptable range for the control room envelope. This requirement is consistent with the present requirements for bottled air systems.

- The regulatory treatment of the portable air supply and the non-safety-grade ventilation system will be in accordance with the staff's position described in Section A of this paper.

The staff agrees with EPRI's concept of the safety-grade pressurization system and the use of safety-grade connections for the pressurization system to allow the use of backup, portable air supplies after 72 hours to minimize operator doses for the duration of the accident. However, the staff has some reservations about limiting the occupancy inside the control room envelope to 5 people for 72 hours. Each of the passive ALWR designs includes design operational conditions similar to the interim operational conditions allowed at existing plants while they implemented permanent modifications to upgrade the systems to meet the requirements of GDC 19. These interim operational conditions were allowed for only a limited period of time because they may not have ensured sufficient control room habitability for the life of the plant. Therefore, a designer must demonstrate (1) the feasibility and capability of the safety-grade pressurization systems to satisfy GDC 19 criteria regarding control room habitability and (2) the availability and capability of the backup air supplies.

To meet the applicable provisions of GDC 4 and 19, both the passive AP600 and SBWR designs provide a safety-related pressurization system to maintain at least 31.1 Pascal (1/8-inch water gauge (WG)) positive differential pressure. The AP600 and SBWR designs also claim that unfiltered leakage into the control room envelope will be restricted to 1.4E-4 to 2.4E-4 cubic meters per second (0.3 to 0.5 cubic feet per minute), respectively. The vendor-specific reviews will be based on the guidelines of SRP Section 6.4, including experience obtained from the operating plants concerning (1) the provisions for maintaining and periodically testing for leaktightness to maintain at least 31.1 Pascal positive pressure relative to all surrounding areas, (2) the adequacy of the ESF filtration system, if needed, (3) the ability of the postaccident safety-related cooling to maintain a habitable environment for control room operators and to provide equipment operability, and (4) protection against the effects of accidental release of toxic gases and smoke inside the control room pressure boundary.

Each of the passive ALWR designs includes non-safety ventilation systems for the control room envelope. The system would be switched to a recirculation mode with filtered makeup on high radiation signal and would be available for control room habitability as long as the ac power is available and the system is operational. The non-safety system is isolated from the control room on a high-high radiation signal measured in the HVAC duct supplied from the non-safety system. There is some probability that the non-safety HVAC systems would be available for control room habitability during a postulated design-basis accident in a period when ac power is available. However, this system and the power supplies are non-safety-related, as designed, and cannot be relied upon for control room habitability during a postulated design-basis accident. Therefore, no credit for the non-safety system can be taken in the safety analysis for design-basis accidents.

The staff will separately consider the control room habitability of each vendor's design for acceptance. The staff will review the designs for control room habitability to ensure that the requirements specified in GDC 19 are met and that personnel and equipment in the control room have a suitable environment for the duration of the accident.

The staff recommends that the Commission approve the following positions on control room habitability for passive plants:

1. The concept of using a passive, safety-grade control room pressurization system which would use bottled air to keep operator doses within the limits of GDC 19 and SRP 6.4, Revision 2 of the SRP for the first 72 hours of the event, and safety-grade connections for the pressurization system to allow the use of offsite, portable air supplies is acceptable if needed after 72 hours to minimize operator doses for the duration of the accident.

2. COL holders must demonstrate through performance of the applicable ITAAC, the feasibility and capability of a pressurization system and the capability and availability of backup air supplies to maintain control room habitability for the duration of the accident.

3. The regulatory treatment of the portable air supply and the non-safety-grade ventilation system should be in accordance with the staff's position described in Section A of this paper.

**E. Reliability Assurance Program**

In SECY-89-013, "Design Requirements Related to the Evolutionary ALWR," the staff stated that the reliability assurance program (RAP) would be required for design certification to ensure that the design reliability of safety-significant SSCs is maintained over the life of a plant. The staff had informed the ALWR vendors and EPRI that it was considering this matter in November 1988.

The ALWR RAP would apply to those plant SSCs that are risk-significant (or significant contributors to plant safety) as determined by using probabilistic, deterministic, or other methods of analysis used to identify and quantify

risk such as the design certification PRA. The purposes of the RAP are to provide reasonable assurance that (1) an ALWR is designed, constructed, and operated in a manner that is consistent with the reliability assumptions and risk insights for these risk-significant SSCs, (2) the reliability of these risk-significant SSCs does not degrade during plant operations, (3) the frequency of transients that challenge ALWR SSCs are minimized, and (4) these SSCs function reliably when challenged.

The staff views the RAP for ALWRs as a two-stage program. The first stage applies to the design phase of the plant life cycle, and would be referred to as the design reliability assurance program (D-RAP). The second stage applies to the construction and operations phases of the plant life cycle, and would be referred to as the operational reliability assurance program (O-RAP). An applicant for design certification would be required to establish the scope, purpose, objective, and essential elements of an effective RAP and would implement those portions of the D-RAP that apply to design certification. A combined license (COL) applicant will be responsible for augmenting and completing the remainder of the D-RAP to include any site-specific design information. Once the D-RAP has been established and the risk-significant SSCs identified and prioritized, the procurement, fabrication, construction, operation, and maintenance of these SSCs would be accomplished under the licensee's O-RAP.

The O-RAP can be thought of as an inclusive program that integrates aspects of existing programs (e.g., maintenance, surveillance testing, inservice inspection, inservice testing, and quality assurance) to achieve its objective. The O-RAP would apply to the construction and operation phases of plant life. Reliability performance goals for risk-significant SSCs would be established under the O-RAP, based on information from the D-RAP. The COL applicant would establish performance and condition monitoring requirements to provide reasonable assurance that the reliability of risk-significant SSCs is maintained or not unacceptably degraded. However, the RAP does not attempt to statistically verify the numerical values used in the PRA through performance monitoring. In addition, O-RAP would provide a feedback mechanism for periodically re-evaluating risk significance based on actual equipment, train, or system performance. Most of the O-RAP would be based on the requirements of the Maintenance Rule, 10 CFR 50.65, whose scope includes systems, structures, and components that are: (1) safety-related and (2) non-safety-related (a) relied upon to mitigate accidents or transients or used in plant emergency operating procedures; or (b) whose failure could prevent safety-related structures, systems, and components from fulfilling their safety-related function or (c) whose failure could cause a reactor scram or actuation of a safety-related system.

The staff and the ACRS have discussed the form and content of the ALWR RAP. In letters and during meetings with the staff, the ACRS noted the similarity between the Maintenance Rule, the license renewal rule, and the RAP. The ACRS has stated that the staff should issue consistent guidance on the elements of an acceptable program that will satisfy these three sets of requirements. In separate correspondence, the staff has provided the following discussion responding to the ACRS comments.

Implementation of the Maintenance Rule following the guidance contained in RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," will meet the requirements of the O-RAP for degradation in SSC reliability or availability associated with maintenance. SSCs which are risk-significant (i.e., those within the scope of O-RAP) are given special treatment during implementation of the maintenance rule. They may be either monitored against specific goals or subject to preventive maintenance which assures acceptable performance and requires root cause analysis and corrective action for failure to meet performance criteria. Based upon industry guidance in NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," which is endorsed by RG 1.160, performance criteria for risk significant SSCs will include consideration of overall SSC availability. If an SSC failure occurs, the licensee will be required to determine whether or not it was maintenance preventable. Where failures are determined to be maintenance preventable, corrective actions and an evaluation of the effectiveness of that action on subsequent performance must be taken. Where failures of safety-related SSCs are caused by design deficiencies or operational errors, the quality assurance (QA) requirements of 10 CFR Part 50 Appendix B require corrective actions.

Therefore, implementation of the Maintenance Rule consistent with RG 1.160 plus corrective action for design or operational error-related failures under Appendix B QA programs, would meet the requirements for O-RAP for risk-significant, safety-related SSCs. Corrective action for design errors or operational errors which degrade non-safety, risk significant SSCs would require corrective action pursuant to O-RAP. Maintenance preventable failures for the SSCs would be evaluated and corrected pursuant to the Maintenance Rule. Thus, the difference between Maintenance Rule implementation and O-RAP relates to treatment of risk-significant non-safety SSCs whose failure is due to design or operational error.

The objective of an O-RAP is to provide reasonable assurance that the reliability and availability of SSCs are maintained commensurate with their risk-significance. The staff believes that this objective could be achieved through implementation of reliability performance monitoring, problem and failure identification, and a comprehensive corrective action program. The O-RAP corrective action program would include performance of a detailed root cause analysis of failures of risk-significant SSCs, implementation of effective corrective actions taken in response to all failures, and verification that the corrective action implemented was effective.

Staff Position on RAP

The staff's position is, for design certification of all ALWRs, a RAP applicable to design certification (D-RAP) should be required, and for a COL application that references a certified design, a RAP plan (augmented D-RAP and O-RAP) and inspections, tests, analyses, and acceptance criteria (ITAAC) should be required. The SSAR should include the details of the D-RAP, including the conceptual framework, program structure, and essential elements. The SSAR for the D-RAP should also (1) identify, prioritize, and list the risk-significant SSCs based on the design certification PRA, deterministic methods, such as, but not limited to, nuclear plant operating experience and relevant

component failure data bases; (2) ensure that the design certification applicant's design organization determines that significant design assumptions, such as equipment reliability and unavailability, are realistic and achievable; (3) include design assumption information for the equipment procurement process; and (4) provide these design assumptions to the COL for consideration in the O-RAP. A COL applicant would augment the design certification D-RAP with site-specific design information and would implement the balance of the D-RAP, including information for the procurement process. The COL applicant would also establish and implement the O-RAP. A COL applicant would be required to submit a RAP plan that integrates the design certification D-RAP, site-specific design information and augmented D-RAP, including information for the procurement process, and the O-RAP.

The O-RAP should consist of reliability performance monitoring, problem and failure identification, root cause analyses, and a corrective action program. However, the RAP does not attempt to statistically verify the numerical values used in the PRA through performance monitoring. The O-RAP corrective action program should include performance of a detailed root cause analysis of all failures of risk-significant SSCs, implementation of effective corrective actions taken in response to all failures, and verification that the corrective actions were effective.

Any SSCs identified as risk-significant in the D-RAP, by actual performance during operation or other methods, would require performance monitoring under the O-RAP. The reliability performance monitoring of risk-significant SSCs under O-RAP would be similar to that required by the Maintenance Rule (10 CFR 50.65). The performance targets or goals established and used with the reliability performance monitoring should provide a means to identify problems and equipment degradation prior to failure. Root cause analyses in the O-RAP would be required for each failure of a risk-significant SSC. Also, corrective actions taken in response to failures or problems and the results of those corrective actions would be monitored as part of the O-RAP.

The O-RAP should also make use of SSC data generated as part of the implementation of existing requirements and programs. For example, results from surveillance testing, inservice inspection and testing, and quality assurance activities, could provide a means of obtaining information on performance and reliability of risk-significant SSCs during procurement, construction, fabrication, testing, operation, and maintenance.

The COL applicant's RAP plan that covers the augmented D-RAP and O-RAP would be reviewed and approved by the NRC staff at the time the COL is issued, with all subsequent changes subject to NRC staff approval prior to implementation, similar to current QA Programs. The staff would verify implementation of the RAP plan with inspections and audits during detailed design, procurement, fabrication, construction, and testing prior to fuel load and would continue to inspect and audit implementation of the reliability assurance program for the duration of the license.

In accordance with SECY-92-287, the staff is proposing a regulation that requires an application for design certification to include: a description of the reliability assurance program used during the initial design that

includes, scope, purpose, and objectives; the methodology used to evaluate and prioritize the structures, systems, and components in the certified design based on their degree of risk-significance; and a list of the structures, systems, and components designated as risk-significant. For those structures, systems, and components designated as risk-significant, an application for design certification must also include: the methodology used to determine dominant failure modes that considered industry experience, analytical models, and existing requirements; and key reliability assumptions and risk insights from the PRA including any operation, maintenance, and monitoring activities that should be considered by a licensee that references the standard design.

The staff is also proposing a regulation that would require each application for a combined license that references a certified design would be required to include: a proposed reliability assurance program plan, applicable for the entire life of the plant, that incorporates the RAP from that certified design; and proposed tests, inspections, and analyses, and acceptance criteria, as required by 10 CFR 52.79(c), for the reliability assurance program plan. Additionally, each licensee under 10 CFR Part 52 would implement the reliability assurance program plan approved by the NRC.

The staff recommends that the Commission approve the staff's position that requirements concerning reliability assurance be incorporated into the design-specific rulemaking for an applicant for design certification and for an applicant for a combined license, that references a certified design.

**F. Station Blackout**

The station blackout rule (10 CFR 50.63) allows design alternatives to ensure that an operating plant can be safely shut down if all ac power (offsite and onsite) is unavailable. In SECY-90-016, "Evolutionary LWR Certification Issues and Their Relationship to Current Regulatory Requirements," the staff concluded that the preferred method of demonstrating compliance with 10 CFR 50.63 for evolutionary designs is by installing a spare (full-capacity) alternate ac power source of a diverse design.

The passive ALWR designs do not require ac power for 72 hours following an event and will include provisions for offsite assistance (including additional ac power) beyond 72 hours. Thus, EPRI and the passive plant designers have not made the same provisions for certain ac power system features found in existing plants or in the evolutionary plant designs. The passive designs lack an alternate ac power source and a normally available second offsite power circuit. They also use non-safety-grade emergency generators (typically diesel generators on existing plants) and non-safety-grade ac electrical distribution systems. Each of these is addressed below.

An alternate ac power source or the ability to cope with a station blackout for a specified duration are the options available to comply with the requirements of the station blackout rule. The staff prefers the use of an alternate ac power source to meet the requirements of the rule in evolutionary plant designs because it offers several advantages. An alternative ac power source could power a larger complement of shutdown equipment and bring the plant to cold shutdown, it could be used for other purposes in addition to station

blackout, it is not limited by time while providing power during a station blackout, and it provided for a uniform hardware approach requiring less analysis and fewer specialized operating procedures. However, EPRI and the passive plant designers stated that the passive plants will be designed to remain in a safe and stable condition for 72 hours without ac power, and without operator actions. This period can be extended well beyond 72 hours with preplanned offsite assistance and simple operator actions. This strong coping capability, reduced reliance on ac power, and minimal required operator actions would seem to obviate the need for an alternate ac source. However, EPRI also reduced the requirements on certain other ac power system features on which the station blackout requirements were premised.

GDC 17 requires that two offsite power circuits be available during plant operating modes. In the URD for the passive plant designs, however, EPRI required that the design include only a single offsite power circuit to supply the plant loads during operating modes. A second circuit is required in the passive plant designs for use only "in the event of an extended unavailability of the normal power supply, e.g., during plant outages." In the passive URD, EPRI stated as rationale for this requirement that it

> will ensure that adequate power supply will be maintained (either from another offsite source at the same site or from offsite) at all times during plant shutdown modes when major maintenance is required on one of the onsite power sources or on the normal offsite circuit.

The staff believes that if two offsite circuits are not available during plant operating modes, the frequency of loss-of-offsite power events and the time needed to recover offsite power will likely be greater than they are for existing plants. The designer should evaluate these difficulties against the stronger coping capability of the passive plant designs. The passive URD also requires that installed spare main and auxiliary transformers be available to replace their counterparts in no more than 12 hours, which should help to reduce the likelihood of an extended loss of the single normally available offsite power circuit.

In addition, EPRI and the passive plant designers are providing non-safety-grade onsite emergency generators (diesel generators or combustion turbine generators) and non-safety-grade ac electrical distribution systems. The staff believes that at least two aspects of this approach could directly affect station blackout. EPRI specified an overall reliability of 0.9 for the emergency generators. The maintenance unavailability and the start/run reliability that EPRI indicates would be consistent with this overall reliability are worse than typically seen on safety-grade diesel generators in existing plants. Secondly, EPRI stated that the emergency generators could be used as peaking units to supply power to the grid. EPRI and the passive plant designers, however, have not provided for a distribution system design that would facilitate the use of the emergency generator in this manner, since it would require that the power be delivered to the grid through the plant buses and distribution circuits. Both of the foregoing provisions could increase the likelihood of a station blackout.

Each of the ac power system features discussed in this section shares two aspects. They are viewed as non-safety systems or components for the passive plant designs, and their potential negative effects on station blackout must be judged against the strong coping capability of the passive plants. The staff, therefore, concludes that this issue is a good candidate to be addressed by the process for the regulatory treatment of non-safety systems described in Section A of this paper.

The staff recommends that the Commission approve the staff's proposal to resolve the station blackout issue and related GDC 17 issues on passive ALWR designs by evaluating the ac power system features discussed above under the process defined herein for resolving the regulatory treatment of non-safety systems issue. The staff will pursue regulatory treatment of these features if they are found to be risk significant or are relied on to meet the R/A missions.

**G. Electrical Distribution**

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," March 25, 1992, the staff recommended that the Commission approve its position that an evolutionary plant design should include the following elements:

- an alternate source of power to the non-safety loads unless the designer 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 plants

- 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 the staff requirements memorandum (SRM) of August 15, 1991, the Commission approved the staff's positions. In a letter of May 5, 1992, EPRI stated that this issue does not apply to passive designs.

The first position identified above involved the lack of a second source of power on evolutionary plant designs (typically an offsite circuit on existing plants) to the traditional non-safety electrical buses that power plant loads required for unit operation. These loads include the reactor coolant pumps (recirculation pumps for BWRs), feedwater pumps, condensate pumps, and circulating water pumps. In SECY-91-078, the staff took this position to ensure that a second power source be provided to a sufficient string of these traditional non-safety loads so that forced circulation could be maintained, and the operator would have the normal complement of non-safety equipment available to bring the plant to a stable shutdown condition after a loss of the normal power supply and plant trip.

In the passive plant designs, the same complement of loads identified above (with the exception of the recirculation pumps in the BWRs that are no longer

used) are fed from traditional non-safety load buses with only a single source of offsite power available to them. However, recognizing the strong coping capability without ac power of the passive plant designs, EPRI has not required that a second offsite power source normally be available to any of the plant loads, non-safety or safety.

The staff took the second position in SECY-91-078 to address the connection of at least one offsite circuit directly to the safety buses with no intervening non-safety buses. In the evolutionary designs, this was accomplished with a direct connection of the second offsite circuit to the safety-grade diesel generator buses. The configuration shown in the passive URD is similar to that for the evolutionary plant, except that the second circuit is only intended to be available during extended plant outages as a maintenance type feed. Furthermore, the diesel generator buses to which the second circuit is connected and most of the ac distribution system are non-safety-grade. Thus, intervening non-safety buses and one transformer are located between the second circuit and the safety-grade ac bus that is now located at the 480-volt motor control center level. The one normally available offsite power circuit connection to the safety buses also has a number of intervening non-safety buses and transformers.

Both of the positions on this issue are closely tied to the lack of a second normally available offsite circuit identified in Section F of this paper. The staff, therefore, concludes that this issue is a good candidate to be addressed by the process for the regulatory treatment of non-safety systems described in Section A of this paper.

The staff recommends that the Commission approve the staff's proposal to resolve the electrical distribution issue on passive ALWR designs by evaluating the ac power system features using the process defined herein for resolving the regulatory treatment of non-safety systems. The staff will pursue regulatory treatment of these features if they are found to be risk significant or are relied on to meet the R/A missions.

**H. Inservice Testing of Pumps and Valves**

In SECY-90-016, the staff recommended that the Commission approve the following four issues for the inservice testing of safety-related pumps and valves beyond the current regulatory requirements in 10 CFR 50.55(a) for 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 to disassemble and inspect pumps and valves to detect unacceptable degradation that cannot be detected through the use of advanced, nonintrusive techniques.

The staff concluded that these requirements are necessary to give adequate assurance of operability of the components.

In its SRM of June 26, 1990, the Commission approved the staff's position as supplemented in the April 27, 1990, staff response to ACRS comments. In that response, the staff agreed with the ACRS recommendations to emphasize the requirements of Generic Letter (GL) 89-10 for evolutionary plants, to resolve check valve testing and surveillance issues, and to indicate how these requirements are to be applied to evolutionary plants. The staff also agreed that the requirements should permit consideration of proposed alternatives for meeting inservice and surveillance requirements. The Commission further noted that due consideration should be given to the practicality of designing testing capability, particularly for large pumps and valves.

The staff will conduct its plant-specific reviews with consideration that SECY-90-016 guidelines on design for testing at design basis conditions may not be practical in all cases, particularly for large pumps and valves. The staff is requesting that a qualification test (under design basis differential pressure) be conducted before installation and inservice valve testing be conducted under the maximum practicable differential pressure and flow when it is not practicable to achieve design basis differential pressure during an inservice test.

In its letter of May 5, 1992, EPRI stated that the ALWR program agrees with the above staff positions for the passive and evolutionary plants. In its letter of August 17, 1992, the ACRS supported the staff's recommendation that the above design, testing, and inspection provisions should be imposed on all safety-related pumps and valves for passive ALWRs.

The staff recommended that the Commission approve the position that these requirements should be imposed on passive ALWRs. The staff also concluded that additional inservice testing requirements may be necessary for certain pumps and valves in passive plant designs. The unique passive plant design places significant reliance on passive safety systems, but also depends on non-safety systems (which are traditional safety-related systems in current LWRs) to prevent challenges to passive systems. Therefore, the reliable performance of individual components is a very significant factor in enhancing the safety of passive plant design. The staff recommends that the following provisions be applied to passive ALWR plants to ensure reliable component performance:

1. The staff may not require important non-safety-related components to meet criteria similar to safety-grade criteria. However, the important non-safety-related pumps and valves as identified by the RTNSS process should be designed to accommodate testing in accordance with ASME Code, Section XI, requirements. Specific positions on the inservice testing

requirements for those components will be finalized when the staff completes its review of the regulatory treatment of non-safety systems.

2. ASME/ANSI OM Part 10 referenced in Section XI, ASME Code, 1989 Edition, provides for the relaxation in the valve testing frequency from quarterly intervals to cold shutdowns or refueling outages if testing during normal plant operations or cold shutdown conditions is not practical. These rules do not accommodate quarterly testing because they address the testing of valves in currently operating reactors where the detailed piping system designs were completed before the NRC promulgated the inservice testing requirements. The vendors for advanced passive reactors, for which the final designs are not complete, have sufficient time to include provisions in their piping system designs to allow testing at power. Quarterly testing is the base testing frequency in the Code and the original intent of the Code. Also, the COL holder may need to test more frequently than during cold shutdowns or at every refueling outage to ensure that the reliable performance of components is commensurate with the importance of the safety functions to be performed and with system reliability goals. Therefore, to the extent practical, the passive ALWR piping systems should be designed to accommodate the applicable Code requirements for the quarterly testing of valves, rather than to allow designs that only accommodate testing during cold shutdowns or refueling outages.

3. The passive system designs should incorporate provisions (1) to permit all critical check valves to be tested for performance in both forward and reverse flow directions and (2) to verify the movement of each check valve's obturator during inservice testing by observing the direct instrumentation indication of the valve position such as a position indicator or by performing nonintrusive test methods.

4. The passive system designs should incorporate provisions to test safety-related power-operated valves under design basis differential pressure and flow. The design basis capability of these types of valves should be verified before the valves are installed, before startup, and periodically through a program similar to that recommended for motor-operated valves in GL 89-10. The staff will determine if and the extent to which this concept should be applied to power-operated valves in important non-safety-related systems when the staff completes its review of the regulatory treatment of non-safety systems.

5. To the extent practical, provisions should be incorporated to verify that motor-operated valves (MOVs) in a safety-related system are capable of recovering from mispositioning. Mispositioning may occur through actions taken locally (manual or electrical), at a motor control center, or in the control room, and includes deliberate changes of valve position to perform surveillance testing. The staff will determine if and the extent to which this concept should be applied to MOVs in important non-safety-related systems when the staff completes its review of the regulatory treatment of non-safety systems.

STAFF REQUIREMENTS MEMORANDUM
ON SECY-94-084
"POLICY AND TECHNICAL ISSUES
ASSOCIATED WITH THE REGULATORY
TREATMENT OF NON-SAFETY SYSTEMS
AND COMSECY-94-024 - IMPLEMENTATION
OF DESIGN CERTIFICATION AND
LIGHT-WATER REACTOR DESIGN
ISSUES"
JUNE 30, 1994

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICE OF THE
SECRETARY

June 30, 1994

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

FROM: John C. Hoyle, Acting Secretary

SUBJECT: SECY-94-084 - POLICY AND TECHNICAL ISSUES ASSOCIATED WITH THE REGULATORY TREATMENT OF NON-SAFETY SYSTEMS
and
COMSECY-94-024 - IMPLEMENTATION OF DESIGN CERTIFICATION AND LIGHT-WATER REACTOR DESIGN ISSUES

The Commission has made the following determinations on the staff's recommendations in SECY-94-084:

A. Regulatory Treatment of Non-Safety Systems

The Commission (with all Commissioners agreeing) has approved the staff's recommendation on RTNSS. However, the Westinghouse comments on this item, as stated in the Attachment to NTD-NRC-94-4145 should be accommodated.

The Chairman believes that the licensees should use the complete plant PRA as opposed to the "focused PRA" to provide an integrated assessment of the relative importance of various systems and components. The focused PRA model does not include some non-safety systems whose performance would affect the calculated risk contribution. Other methods, perhaps utilizing risk importance measures, could be identified which still incorporate the information of the complete PRA. The Chairman requested that the staff evaluate this approach.

B. Definition of Passive Failure

The Commission (with all Commissioners agreeing) has approved the staff's recommendation on this item.

SECY NOTE: SECY-94-084 WAS RELEASED TO THE PUBLIC ON APRIL 7, 1994. THIS SRM AND THE VOTE SHEETS OF ALL COMMISSIONERS WILL BE MADE PUBLICLY AVAILABLE 10 WORKING DAYS FROM THE DATE OF THIS SRM.

Commissioner de Planque cautioned that in some situations, a design that considers such failures may overall be less reliable (due to added complexity, new failure modes) than one where the valve is treated as passive.

C. Safe Shutdown Requirements

The Commission (with all Commissioners agreeing) has approved the staff's recommendation on this item. With respect to the 72-hour capacity of the passive RHR system water pool, the requirements for replenishing the water in the pool should be based on design-specific attributes, and the justification presented by the applicant should not be based solely on the URD 72-hour criterion. The staff should be receptive to arguments for longer periods, if technically justified.

D. Control Room Habitability

The Commission (with all Commissioners agreeing) has decided to defer decision on this issue until the staff and the applicant can discuss the issue at greater length to resolve whether to require testing of the leak tightness of the control room at every reload outage. When the staff returns with a recommendation, it should also address whether, and if so, how the control room should be manned during the 72-hour testing proposed for each refueling outage.

E. Reliability Assurance Program (RAP)

The Commission (with all Commissioners agreeing) is in agreement with the general purpose of the RAP. The staff should modify the statement of purpose to read as follows:

> "...to provide reasonable assurance that (1) an ALWR is designed, constructed, and operated in a manner that is consistent with the ~~reliability~~ assumptions and risk insights for these risk-significant SSCs, (2) the ~~reliability of these~~ risk-significant SSCs do not degrade to an unacceptable level during plant operations, (3) the frequency of transients that challenge ALWR SSCs are minimized, and (4) these SSCs function reliably when challenged."

The requirement for the D-RAP should be monitored within the bounds of the Commission's Safety Goals Policy, including the approved subsidiary objectives. The staff should address how it will monitor licensees' reliability assurance efforts without effectively translating industry design reliability assumptions into new regulatory requirements which result in CDF and CCFP values that are lower than the subsidiary objectives which the Commission approved for use.

(References: SRM on SECY-89-311, December 15, 1989; SRM on SECY-89-102, June 15, 1990; and SRM on SECY-90-016, June 26, 1990).

The Commission (with all Commissioners agreeing) has approved D-RAP subject to the resolution of the OGC recommendation to implement the D-RAP using the ITAAC process.

The Commission (with all Commissioners agreeing) has disapproved the staff's proposal to require that an O-RAP be continued for the life of the COL license. The staff should ensure that the objectives of the O-RAP are incorporated into existing programs for maintenance or quality assurance.

F. Station Blackout

The Commission (with all Commissioners agreeing) has approved the staff's recommendation on this item.

G. Electrical Distribution

The Commission (with all Commissioners agreeing) has approved the staff's recommendation on this item.

H. Inservice Testing of Pumps and Valve

The Commission (with all Commissioners agreeing) has deferred a decision on staff's recommendation on this item. The staff should provide additional explanation on why it is necessary to require testing in both directions for valves whose safety mission is in one direction only. It is also not clear whether the blowdown valves will be required to be periodically tested at design-basis conditions, or whether the staff will accept validation through "type" testing prior to installation. For example, in Italy, Westinghouse is testing several valves under design-basis conditions. Will the staff accept these qualification tests as demonstrating the ability of the valves to fulfill their mission, or will periodic testing be required? The requirements for quarterly testing during operation should be determined from a risk perspective, e.g. balancing the risks of testing against the benefits of a simpler design (that might result if testing during operation is not to be done). (See also comment on B above). Valves should be tested under design basis differential pressure and flow during power operation only if the benefits of the test outweigh the potential risk. These issues require further discussion prior to a Commission decision.

The Commission has made the following determinations on the staff's recommendations in COMSECY-94-024:

A. Level of Design Detail Including DAC

The Commission (with all Commissioners agreeing) has approved the staff's recommendation to use of design acceptance criteria (DAC) in the certification process. The Commission assumes and expects the staff to obtain all the information it requires to render its safety decisions.

B. The Two-Tiered Design Certification Rule Structure

The Commission (with all Commissioners agreeing) has approved the staff's recommendation for a two-tiered design certification rule structure.

C. The Tier 2* Information Category

The Commission (with all Commissioners agreeing) has approved the staff's use of a Tier 2* information category to identify certain Tier 2 information that would require prior NRC approval before it could be changed by a COL applicant or licensee.

D. The Reliability Assurance Program

This issue is addressed in item E. above.

cc: The Chairman
Commissioner Rogers
Commissioner Remick
Commissioner de Planque
OGC
OCA
OIG
Office Directors, Regions, ACRS, ACNW, ASLBP (via E-Mail)

# ANNEX C

# DOCUMENTS RELATED TO DRAFT COMMISSION PAPER, "SOURCE TERM RELATED TECHNICAL AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND PASSIVE LIGHT-WATER-REACTOR DESIGNS" FEBRUARY 10, 1994

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February 10, 1994

MEMORANDUM FOR: The Chairman
Commissioner Rogers
Commissioner Remick
Commissioner de Planque

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: DRAFT COMMISSION PAPER, "SOURCE TERM RELATED TECHNICAL AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND PASSIVE LIGHT-WATER-REACTOR DESIGNS"

The staff has identified several technical and licensing issues pertaining to the application of the revised accident source terms to evolutionary and passive light-water reactor designs. These issues are discussed in the enclosed draft Commission paper, "Source Term Related Technical and Licensing Issues Pertaining to Evolutionary and Passive Light-Water-Reactor Designs."

The staff proposes to release this draft Commission paper to the public 3 days after the date of this memorandum. Although certain topics presented in the enclosed paper are the subject of continuing staff and the Office of the General Counsel discussions, release of this draft paper will enable effective interaction with the Advisory Committee on Reactor Safeguards and industry on these issues. The staff will forward a final Commission paper after consideration of comments.

Original signed by
James M. Taylor

James M. Taylor
Executive Director
for Operations

Enclosures: As stated

cc: SECY
OGC
OCA
OPA
ACRS

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Ashok Thadani, NRR
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DRAFT

**FOR:** The Commissioners

**FROM:** James M. Taylor
Executive Director for Operations

**SUBJECT:** SOURCE TERM-RELATED TECHNICAL AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND PASSIVE LIGHT-WATER-REACTOR DESIGNS

**PURPOSE:**

To inform the Commission of staff positions pertaining to source term-related technical and licensing issues, proposed review plan, and departures from previous practices for evolutionary and passive light-water-reactor (LWR) design certification reviews.

**BACKGROUND:**

In SECY-90-016, "Evolutionary Light-Water-Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," the staff recommended that the Commission approve for evolutionary LWRs, among other things, that the staff consider any deviations from current methodology used to calculate 10 CFR Part 100 doses on a case-by-case basis using engineering judgment, including updated information on source term and equipment reliability. The Commission approved this recommendation in a staff requirements memorandum dated June 26, 1990.

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In SECY-92-127, "Revised Accident Source Terms for Light-Water Nuclear Power Plants," the staff provided the Commission a draft report (later published as draft NUREG-1465) describing revised accident source terms for light-water nuclear power plants to replace those of Technical Information Document (TID) -14844, published in 1962. The staff also described the process it would use to finalize the report, including both a public comment and peer review period. Any specific applications of the new source term prior to the completion of this process were to be submitted to the Commission for preliminary review.

In this paper, the staff notes that the review process for NUREG-1465 is essentially complete and presents its positions on (1) the closure of source term-related open issues in its FSERs for the EPRI requirements documents for evolutionary and passive plant designs and (2) the generic implementation of source term-related issues in evolutionary and passive LWR design certification reviews. The staff is not requesting a preliminary review by the Commission of these positions at this time because they are primarily technical applications of previous Commission policy decisions. The application of the revised source terms is reactor type and design specific and, therefore, the final staff positions will be addressed on a design-specific basis for Commission approval in the staff's final safety evaluation reports (FSERs) for each evolutionary [Asea Brown Boveri-Combustion Engineering (ABB-CE) System 80+] and passive LWR design.

**DISCUSSION:**

The current reactor accident source terms for fission-product release from the reactor core into the containment are set forth in Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," and Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors." These source terms were derived from TID-14844. The regulatory guide source terms are used in conjunction with postulated design-basis accidents (DBAs). Revised reactor accident source terms have been proposed in draft NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants." These source terms were derived from examination of a set of accident sequences for current LWR designs and reflect the current understanding of severe accidents and fission-product behavior.

Consistent with a previous Commission directive regarding the need to develop revised accident source terms, the staff is proposing that such source terms be used primarily in radiological consequence assessments in the following areas of evolutionary and passive LWR design certification reviews:

(1) equipment qualification
(2) control room habitability
(3) engineered safety features atmosphere cleanup systems
(4) primary containment leak rate
(5) containment isolation (radiological aspect)
(6) post accident sampling
(7) shielding and vital area access

In this paper the staff identifies its significant technical positions relative to the implementation of the revised source terms for evolutionary and passive LWR designs. These positions are more fully discussed in Enclosure 1. On the basis of these positions, the staff believes that it can close out all source term-related open issues in its FSERs for the EPRI requirements document for evolutionary and passive plant designs and also the open issues in the evolutionary and passive LWR design reviews. The staff has not conducted a detailed evaluation of how to apply the revised source term information to existing plants.

The revised source term is not proposed as a requirement for existing plants. Preliminary indications are that the revised release estimates, in conjunction with improved insights on timing and best estimate credit for fission product removal mechanisms, would not result in more stringent requirements if applied to existing plants, and may support relaxation of certain present regulatory requirements, particularly those associated with time of appearance. Because it reflects our current understanding of source term behavior, the staff plans to make the revised source term information available to existing licensees for voluntary proposals to modify current requirements that may be overly conservative. However, due to the large number of design and operational areas impacted by the source term and the plant specific nature of these impacts, the staff has not determined at this time the extent to which these insights could be implemented for existing plants. The staff is developing an overall approach to application of NUREG-1465 source term to currently operating reactors and expects to forward its recommendations to the Commission within 60 days following the final NUREG-1465 publication.

In Enclosure 1, the staff discusses 12 source term-related policy, technical, and licensing issues pertaining to either evolutionary LWRs, passive LWRs, or both. The staff previously identified all of these issues in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water (ALWR) Designs," dated April 2, 1993. Of these 12, the staff highlights *(in italics)* the 5 most significant issues (Issues 1 through 5) in the following summary discussion.

Draft NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants"

In SECY-92-127, the staff provided, for the Commission's information, a draft staff report containing proposed revised accident source terms for light-water nuclear power plants to replace those of Regulatory Guides 1.3 and 1.4. The draft report was issued for public comment in July 1992 as NUREG-1465. Draft NUREG-1465 addressed the current understanding of LWR severe accident and fission-product behavior, including the quantity and chemical forms of fission products released into the reactor containment following a severe reactor accident as well as fission-product release timing. Since issuing draft NUREG-1465, the staff has received more than 200 comments from 20 U.S. and foreign organizations.

The more significant public comments concern (1) the release fractions of the less volatile and nonvolatile elements such as barium, strontium, and cerium during the early-in-vessel and ex-vessel release phases, (2) chemical forms of

iodine, and (3) fission-product release timing. The staff's release fractions of the less volatile and nonvolatile elements are about 10 to 100 times higher than those proposed by EPRI. Draft NUREG-1465 does not address the expected amounts of iodine in organic form in the containment following a DBA, while EPRI specifies 0.15 percent of the core inventory iodine as the organic form.

The staff found in its radiological consequence assessments that the less volatile and nonvolatile elements play a less significant role, contributing less than a few percent of the overall radiological consequences resulting from a DBA, compared to the other major nuclides such as noble gases, iodine, and cesium. Therefore, the staff believes that the potential revisions to the release fractions of the less volatile and nonvolatile elements in draft NUREG-1465, if any, will not materially change the ongoing staff reviews of the evolutionary and passive LWR designs.

For the chemical forms of iodine, the staff found that iodine in organic form may contribute significant offsite radiological consequences for the entire assumed duration of an accident (30 days). As discussed in the Design Basis Accident Assessment section, the staff proposes to assume in its reviews that 0.25 percent of the core iodine inventory is in organic form and that the entire organic fraction will be available for leakage from the containment to the environment following a DBA.

Most of the remaining public comments are largely favorable regarding draft NUREG-1465. NRC contractors are reviewing the comments received and are expected to complete the review within the next several months. The staff expects to issue the final version of NUREG-1465 in 1994. The staff briefed the Commission on the status of the proposed source term update, including the discussion of major comments received on the draft NUREG-1465, on August 3, 1993.

Draft NUREG-1465 also contained a section on mechanisms for removing fission products from the reactor containment atmosphere following a severe reactor accident that was based on the realistic estimates of the natural processes as used in the NUREG-1150 ("Severe Accident Risks: An Assessment for 5 U.S. Nuclear Power Plants") analyses. To obtain more comprehensive information on removal mechanisms to be used in the licensing reviews, the staff is continuing to explore this area with NRC contractor assistance.

The staff has developed its positions on source term-related issues without waiting for more comprehensive information so that it can complete its review of the EPRI requirements documents for evolutionary and passive plant designs and continue the licensing review of the evolutionary and passive LWR designs in order to support the integrated review milestones delineated in SECY-93-097, "Integrated Review Schedules for the Evolutionary and Advanced Light-Water-Reactor Projects." Because current regulatory guides and the Standard Review Plan (SRP, NUREG-0800) give no guidance on how to implement the revised source terms, the staff proposes adopting an interim design-specific approach.

The staff's positions on the source term issues discussed in this paper are based on NRC contractor findings in NUREG/CR reports, preliminary findings in

NRC contractor reports, extensive consultation with these contractors, the staff's engineering judgments, and the need to maintain an appropriate margin of safety.

Design-Basis-Accident Assessment

To evaluate the evolutionary and passive LWR submittals, the staff will utilize the current insights from source term research as described in draft NUREG-1465 regarding fission-product releases into the containment. In determining the effects of such removal mechanisms as sprays, filters, plateout, and aerosol deposition, the staff will use the best available information, including engineering judgment, for the applicable parameters.

The use of physically based source terms as described in draft NUREG-1465 constitutes a major departure from the use of the current regulatory guide source terms. The proposed source terms lead to a qualitatively different fission-product distribution, which is a result of (1) physically based composition, release timing, and rate of releases and (2) inclusion of removal mechanisms based on natural processes.

In Enclosure 2, the staff presents the revised source terms to be used in conjunction with DBAs for reviewing evolutionary and passive LWR designs, including their release timing as well as duration for boiling-water reactors (BWRs) (Table 1) and for pressurized-water reactors (PWRs) (Table 2). The information in these tables, taken from draft NUREG-1465, may be revised when NUREG-1465 is issued in final form. The staff does not expect that any revision will materially change its evaluation or conclusions pertaining to these designs.

The staff-proposed source term release fractions (magnitudes) in draft NUREG-1465 are compared to those of the regulatory guides in Tables 3 and 4 of Enclosure 2. The staff proposes to use only the gap release and the early in-vessel release for design-basis calculations for the evolutionary and passive LWR designs. These source terms encompass a broad range of accident scenarios, including significant levels of core damage with the core remaining in the vessel. These would be the most severe scenarios from which the plant could be expected to return to a safe shutdown condition. Tables 3 and 4 in Enclosure 2 show that the draft NUREG-1465 release fractions associated with these releases are generally comparable to those of TID-14844.

The staff considers the inclusion of the ex-vessel and late in-vessel source terms to be unduly conservative for DBA purposes. Such releases would only result from core damage accidents with vessel failure and core-concrete interactions. For evolutionary and passive LWRs, the estimated frequencies of such scenarios are low enough that they need not be considered credible for the purpose of meeting 10 CFR Part 100.

On the basis of a comparison of the gap and in-vessel releases (excluding ex-vessel releases), the staff finds that the quantities of noble gases and radioiodines are in reasonable agreement, while the proposed new source terms include substantially more of the low volatility solids. For the regulatory

guide source terms, the staff assumed that the fraction of these solids released represents less than 1 percent of the activity available for release, while the revised source terms show significantly larger release fractions for these materials. The important differences are that the proposed new source terms are (1) released over a period of 1.8 (PWR) to 2.5 hours (BWR), rather than instantaneously, and (2) virtually free of the hard-to-remove organic radioiodine. Draft NUREG-1465 did not address organic radioiodine, but the staff believes that the existence of a small amount of organic radioiodine cannot be excluded.

*In radiological assessments of DBAs for evolutionary and passive light-water reactor designs, the staff will:*

*Selectively use the source terms given in draft NUREG-1465 using only "GAP RELEASE" and "EARLY IN-VESSEL RELEASE" (excluding "EX-VESSEL RELEASE" and "LATE IN-VESSEL RELEASE" associated with vessel failure, and core-concrete interaction) in evaluating radiological consequences for DBAs, in evaluating the DBA radiation environmental qualification of electrical and mechanical equipment important to safety, and in evaluating Three Mile Island-related requirements (Issue 1 for System 80+, AP600, and the Simplified BWR (SBWR)).*

*Use the chemical forms of iodine of at least 95 percent cesium iodine as stated in draft NUREG-1465 with 4.75 percent of elemental iodine and hydrogen iodide and 0.25 percent organic iodide (instead of 5 percent elemental iodine and hydrogen iodide as stated in draft NUREG-1465) (Issue 2 for System 80+, AP600, and SBWR).*

Severe Accident Assessment

The staff recognizes that the new source terms have implications for severe accidents as well. The staff's resolution of severe accident issues is based on the inclusion of specific design features to provide assurance of containment integrity for approximately 24 hours following the onset of core damage. Part of the basis for the 24-hour criterion is the fact that natural fission-product removal mechanisms can significantly reduce the suspended aerosol source terms present in the containment within this time period.

Another aspect of the resolution of severe accident issues is the requirement that reasonable assurance exist that equipment needed for severe accident mitigation and sampling will survive in the severe accident environment. This is an issue raised to the Commission in SECY-90-016 and SECY-93-087 and approved in staff requirements memoranda dated June 26, 1990, and July 21, 1993. It is the responsibility of the applicant to demonstrate that all such equipment can survive the radiation environment following a severe accident. As discussed in Enclosure 1, the staff believes that the design-basis source terms described above may not represent reasonable surrogates for the environmental conditions in a severe accident. For a severe accident, the applicant should include the contribution of fission products that may be released ex-vessel. For those reactors that have a reliable ex-vessel

flooding mechanism for the core debris, this contribution will be significantly reduced by aerosol scrubbing.

*In radiological assessments of equipment survivability as a result of a severe reactor accident, the staff will require that the equipment and features needed for severe accident prevention, mitigation, and sampling be designed to provide a reasonable level of confidence that they will operate in a severe accident environment. This environment would include the ex-vessel release, with proper credit for design features to mitigate that release, and the late in-vessel release in addition to the releases for a DBA (Issue 3 for System 80+, AP600, and SBWR).*

Finally, the probabilistic safety assessment (PSA) for each advanced reactor includes an evaluation of the dose consequences associated with a variety of core melt scenarios and containment failure modes. These calculations were performed with the most current understanding of source term behavior. While these assessments generally contain some discussion of uncertainties in fission-product behavior, they do not include any explicit provision for conservatism. As is generally the case in PSAs, these calculations are done on a best-estimate basis.

Design-Specific Reviews and Staff Positions

The staff is reviewing the four advanced reactor designs submitted under 10 CFR Part 52: the General Electric (GE) advanced boiling water reactor (ABWR), the Asea Brown Boveri Combustion Engineering (ABB-CE) System 80+, the Westinghouse AP600 and the GE SBWR.

ABB-CE initially proposed to use the TID-14844 source terms. However, it subsequently submitted to the staff an amendment to its standard safety analysis report (SSAR) incorporating the revised accident source terms as described in draft NUREG-1465.

The System 80+ design provides the same mitigating features for the radiological consequences resulting from a DBA as those provided for the currently operating PWRs (containment spray, engineered safety features (ESF) filtration, etc.), except that safety-grade charcoal adsorbers are not provided. ABB-CE has assumed that the iodine chemical form mainly consists of cesium iodide in particulate form (more than 95 percent). Therefore, the design relies on two high-efficiency particulate air filters (in series) in the ESF filtration system to remove iodine in particulate form with a removal efficiency that is greater than 95 percent. The staff's preliminary review using its positions discussed in this Commission paper showed that the System 80+ design will meet the dose reference values given in 10 CFR Part 100 and the control room operator dose limits given in 10 CFR Part 50.

For the AP600 design, Westinghouse proposed the same accident source terms as those proposed by EPRI for passive plants. The AP600 design provides neither an ESF filtration (e.g., charcoal adsorbers) nor a containment spray system. In SECY-93-087 (Item III.F, Radionuclide Attenuation), the staff informed the Commission that it is still evaluating the need for a containment spray system

for the passive plant designs. Unlike the current generation of operating PWRs, the AP600 design does not include containment sprays. The staff will review the AP600 radiological consequence assessment to determine if the dose reference values in 10 CFR Part 100 and the control room operator dose criteria of General Design Criterion 19 of Appendix A to 10 CFR Part 50 can be met without a containment spray system. That determination will involve evaluation of aerosol removal rates by containment spray, natural deposition, containment leak rates, and post-accident pH control. This issue is addressed in more detail as Issue 9 in Enclosure 1.

GE demonstrated in its SSAR that the ABWR design will meet the offsite dose reference values given in 10 CFR Part 100 using the current TID-14844 source terms. The staff reviewed the ABWR design and performed an independent analysis of the radiological consequences resulting from a postulated DBA and concluded in the FSER that the ABWR design will meet the dose reference values given in 10 CFR Part 100. In its review, the staff accepted the ABWR design without a main steam leakage control system (LCS) that is designed to process main steam leakage through main steam isolation valves following a DBA. The staff also allowed a credit for removal of radioactive iodine in main steam leakage by holdup and plateout in the main steam piping and condenser following a DBA. As discussed in Enclosure 1, these deviations from the provisions of Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants," and the principle of not allowing credit for non-seismically qualified systems are based on the conclusion that these systems are expected to retain sufficient structural integrity to remain effective for fission-product holdup and plateout throughout a DBA.

*The staff will accept the passive BWR plant design without a LCS and allow an appropriate credit for iodine removal in the main steamline and condenser following a DBA (Issue 4 - for SBWR).*

In its SBWR design, GE initially proposed the same accident source terms as those published in NUREG-1465, but recently amended its SSAR to reflect the EPRI accident source terms. The SBWR design provides non-safety-grade containment spray systems without an ESF filtration system. This reduction of safety margin for mitigating radiological consequences may be compensated for by the secondary containment ("safety envelope") design. The safety envelope is a reinforced-concrete structure within the reactor building that forms an envelope completely surrounding the primary containment. It is designed to be capable of periodic testing to ensure its intended performance. The design also relies on lower core power density (lower reactor power) and slower and delayed releases of fission products into the containment.

*The staff will allow appropriate credit for the SBWR safety envelope based on fission-product holdup and decay within this envelope if (1) the vendor specifies that the secondary containment leakage and mixing performance be consistent with the values used by the staff in its radiological assessment and (2) the COL combined license applicant incorporates the secondary containment leakage value specified by the vendor into the plant-specific technical specifications (Issue 5 - for SBWR).*

The staff has essentially completed its review of source term-related issues in the EPRI requirements documents for evolutionary and passive plant designs and in the System 80+ design. It is reviewing the AP600 and SBWR designs. Its positions presented in this paper will form a basis for closing of all source term-related open issues associated with these three designs. The staff does not expect that the publication of NUREG-1465 in final form will materially change its evaluation or conclusions pertaining to these designs.

Margins of Conservatism

The use of the revised source term information is an important departure from previous practice. The new approach which generally yields lower estimates of fission product releases to the environment, will employ a physically-based source term based on substantial research and experience gained over two decades. The TID-14844 non-mechanistic methodology intentionally employed conservative assumptions which were intended to ensure that future plants would provide sufficient safety margins even with the recognized uncertainties associated with accident sequences and equipment reliability. While the TID-14844 source term has served its intended purpose, research started prior to the Three Mile Island Accident provides substantial information on plant behavior under accident conditions. The results of this research are summarized in draft NUREG-1465.

The staff believes that accident research insights and information provided in draft NUREG-1465 and this paper provide a sound, contemporary basis for reactor accident mitigation system designs. While the new information may lead to relaxation in some aspects of the design, it also provides safety benefits by removing unrealistically stringent testing requirements. For example, the TID-14844 source term specifies the instantaneous release of fission-products into the primary containment while reactor accident research insights indicate that releases occur over a period of hours. This change, when used as a partial basis for less stringent closure requirements for fast-acting isolation valves, could result in enhanced reliability and integrity of these valves. Another example includes giving credit for fission-product plateout in the ABWR main steamlines and condenser when no credit has previously been given. This is based on our improved understanding of iodine transport mechanisms and iodine behavior from the revised accident source term research insights. The credit for fission-product plateout in the main steamlines and condenser could also result in enhanced reliability and integrity of the main steam isolation valves, by allowing the staff to use its higher leak rates and less stringent closure requirements.

The NUREG-1465 accident research insights strongly link the revised iodine chemical forms discussed in this paper with the ability to control in-containment pH > 7.0 during the course of an accident. If the proposed plant design features provide for adequate pH control based on a consideration of the various sources of acid additions that could occur during the course of an accident, the staff will not need to consider re-evolution of elemental iodine. Accordingly, there will not be a need to consider accident mitigation features such as containment sprays and charcoal adsorbers. If, however, a particular design does not provide adequate pH control, the staff will require the addition of non-safety grade charcoal adsorbers or a containment spray system.

With considerations such as those described above, the staff believes that the use of the new source terms provides a reasonable basis for establishing acceptable designs.

SUMMARY:

The methodology for implementing the revised source terms gives credit for improved understanding of fission product release timing resulting from accident research as well as more realistic estimates of reductions in the magnitude of potential fission-product releases from containment. These changes will result in reduced requirements for mitigation systems. The staff recognizes this effect and believes that it is justified on the basis of the improved understanding of fission-product behavior. The revised source terms will provide the staff with adequate technical bases for ensuring that safety margin is maintained.

The staff presents its positions on 12 source term-related issues in Enclosure 1. The establishment of staff positions in these areas provides the basis for the staff to: (1) close out all source term-related open issues in the staff's FSERs for the EPRI requirements documents for evolutionary and passive plant designs, (2) complete the ABB-CE System 80+ design certification review, and (3) proceed with its review of the Westinghouse AP600 and the GE SBWR designs.

COORDINATION:

The Office of the General Counsel has reviewed this paper and has no legal objection. This paper will be discussed with the Advisory Committee on

Reactor Safeguards. The staff intends to place a copy of this paper in the Public Document Room three working days from the date of this paper.

James M. Taylor
Executive Director
for Operations

Enclosures:
1. Source Term-Related Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water-Reactor Designs
2. Proposed Reactor Accident Source Terms

DRAFT

ENCLOSURE 1

## SOURCE TERM-RELATED POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND PASSIVE LIGHT-WATER-REACTOR DESIGNS

**ISSUE 1: Selective Use of Accident Source Terms Given in Draft NUREG-1465 (for System 80+, AP600, and SBWR).**

In SECY-90-016, the staff recommended to the Commission that it approve staff consideration of deviations from current methodology used to calculate 10 CFR Part 100 doses on a case-by-case basis using engineering judgment and updated information on source terms and equipment reliability for the evolutionary plant designs. The Commission approved this recommendation in a staff requirements memorandum of June 26, 1990. This section of the Commission paper describes such deviations to selectively use the fission-product release fractions described in NUREG-1465 for radiological consequence assessments of design-basis accidents for the evolutionary and passive light-water-reactor designs.

Present regulations (10 CFR Part 100) require that a fission-product release from the reactor core into the containment as a result of a reactor accident be postulated and that its radiological consequences at the exclusion area boundary and low population zone outer radius be evaluated assuming the "expected demonstrable leak rate" from the containment and the meteorological conditions pertinent to the reactor site, with the implicit assumption that the containment remains intact against the maximum credible accident. Footnote 1 to 10 CFR Part 100 states that the fission-product release to be assumed for Part 100 dose calculations should be "based on a major accident...that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products."

The current source terms are specified in Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," and Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant for Pressurized Water Reactors." The source terms were derived from Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." Although the staff considered the consequences of fission-product release into the containment to represent the bounding reactor accident in terms of credibility, it also evaluated other accident types to verify that these did not result in greater consequences. This practice led the staff eventually to develop and consider a group of accidents, referred to as "design-basis accidents (DBAs)." The DBAs routinely evaluated by the staff as part of the license review include (1) loss-of-coolant accident (LOCA), (2) fuel handling accident, (3) steam generator tube rupture accident (PWR), (4) main steamline break outside

the containment, (5) control rod drop accident (BWR), (6) control rod ejection accident (PWR), (7) failure of small lines carrying primary coolant outside the containment, and (8) spent fuel cask drop accident. The staff used the calculated radiological consequences resulting from these DBAs to ensure that the distances to the exclusion area boundary and the outer radius of the low population zone for a nuclear power plant, in conjunction with the operation of the dose-mitigating engineered safety features (ESF) systems, are sufficient to provide reasonable assurance that the dose reference values in 10 CFR Part 100 will not be exceeded.

For environmental qualification of electrical equipment, 10 CFR 50.49 states that safety-related electrical equipment, certain non-safety-related electrical equipment, and certain post accident monitoring equipment should remain functional during and following design-basis events. Design-basis events are further defined as conditions of normal operation, including anticipated operational occurrences, DBAs, external events, and natural phenomena that the plant must be designed and built to withstand.

In Regulatory Guide 1.89 (Revision 1), "Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants" (June 1984), the staff assumes the TID-14844 source term for determining the radiation environment for qualification of electrical equipment important to safety and for calculating the time integrated radiation doses in the containment and at a point just above the containment sump.

For additional Three Mile Island (TMI)-related requirements, 10 CFR 50.34 refers to the TID-14844 source term (1) to perform radiation and shielding design review, (2) to assess post accident sampling capability, (3) to assess the system design for leakage control and detection outside the containment, and (4) to evaluate control room habitability under accident conditions.

Draft NUREG-1465 lists five fission-product release phases of a severe LWR accident: (1) coolant activity release, (2) gap activity release, (3) early in-vessel release, (4) ex-vessel release, and (5) late in-vessel release. The coolant activity release phase begins with a postulated pipe rupture and ends when the first fuel rod has been estimated to fail. During this phase, the fission products released to the containment atmosphere are those associated with radioactive material in the reactor primary coolant during normal plant operation. Draft NUREG-1465 does not specify fission-product releases to the containment during this phase, since the amounts of the fission products released are small compared with later release phases. The gap activity release phase begins when fuel cladding failure commences and ends when the fuel starts to melt. This phase involves the release of that radioactive material that has collected in the gap between the fuel pellet and cladding.

The early in-vessel release phase begins when fuel and other materials melt and fall to the bottom of the reactor pressure vessel. During this phase, significant quantities of the volatile radionuclides in the core inventory, as well as some of the less volatile radionuclides, are estimated to be released into the containment. This release phase ends when the bottom head of the reactor pressure vessel fails, allowing molten core debris to fall onto the concrete below the reactor pressure vessel.

The ex-vessel release phase begins when the molten core debris exits the reactor pressure vessel and the core-concrete interaction takes place. This phase ends when the debris has cooled sufficiently so that significant quantities of fission products are no longer being released. During this phase, significant quantities of the volatile radionuclides not already released during the early in-vessel phase as well as lesser quantities of nonvolatile radionuclides are released into the containment. Finally, the late in-vessel release phase commences at vessel breach and proceeds simultaneously with the ex-vessel release phase.

The staff proposes to use only the gap release and the early in-vessel release for design-basis calculations for the evolutionary and passive LWR designs. These source terms encompass a broad range of accident scenarios, including significant levels of core damage with the core remaining in the vessel. These would be the most severe scenarios from which the plant could be expected to return to a safe shutdown condition. Tables 3 and 4 in Enclosure 2 show that the draft NUREG-1465 release fractions associated with these releases are generally comparable to those of TID-14844.

The staff considers the inclusion of the ex-vessel and late in-vessel source terms to be unduly conservative for DBA purposes. Such releases would only result from core damage accidents with vessel failure and core-concrete interactions. For evolutionary and passive LWRs, the estimated frequencies of such scenarios are low enough that they need not be considered credible for the purpose of meeting 10 CFR Part 100.

Tables 1 and 2 of Enclosure 2 list the proposed fission-product release fractions of core inventory into the containment, including their release timing and duration. The numerical values in these tables, taken from draft NUREG-1465, may be revised when NUREG-1465 is issued in final form. However, the staff does not expect that the publication of NUREG-1465 in final form will materially change its evaluation or conclusions pertaining to these designs.

*The staff has addressed Issue 1 by selectively using the source terms given in draft NUREG-1465 using only "GAP RELEASE" and "EARLY IN-VESSEL RELEASE" (excluding "EX-VESSEL RELEASE" and "LATE IN-VESSEL RELEASE" associated with vessel failure, and core-concrete interaction) in evaluating radiological consequences for DBAs, in evaluating the DBA radiation environmental qualification of electrical and mechanical equipment important to safety, and in evaluating TMI-related requirements (for System 80+, AP600, and SBWR).*

**ISSUE 2: Iodine Chemical Form (for System 80+, AP600, and SBWR)**

Regulatory Guides 1.3 and 1.4 specify that fission-product releases into the containment consist of 100 percent of the core inventory of noble gases and 50 percent of iodines (half of which are assumed to deposit on interior surfaces of the containment very rapidly). The iodine chemical form is specified to be predominantly elemental iodine (91 percent), with 5 percent assumed to be particulate iodine and the remaining 4 percent assumed to be in the organic form. The 1 percent of "solid" fission products included in TID-14844 was dropped from consideration in Regulatory Guides 1.3 and 1.4.

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

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

EPRI proposed the use of 0.15 percent organic iodide based on 3 percent conversion of elemental iodine to organic iodine (i.e., 5 percent of 3 percent is 0.15 percent). Information from studies by NRC contractors documented in NUREG/CR-4461,- 5732, and- 5950 indicates that iodine and hydrogen iodide may exceed 3 percent. In addition, the staff found in its radiological consequence assessment that iodine in organic form represents a significant offsite dose because no removal mechanisms are available for the organic form of iodine other than decay and passive deposition within the containment.

***The staff has addressed Issue 2 by partitioning radioiodine chemical forms as follows: 95 percent cesium iodide (CsI, or $I^-$ in the aqueous phase); 4.75 percent elemental iodine ($I_2$), and 0.25 percent as organic iodide (e.g., $CH_3I$). The latter two values will be used in preference to a single value of 5 percent $I_2$ recommended in draft NUREG-1465. These values for the iodine chemical form will be used in the System 80+, AP600, and SBWR reviews.***

The evolutionary and passive LWR vendors have stated that they will use the staff's proposed values until final NUREG-1465 is published. The staff will use these chemical forms in evaluating the containment spray removal efficiencies, BWR suppression pool decontamination factors, passive deposition of fission products in the containment, control room habitability, and engineered safety features (ESF) filtration systems in its evolutionary and passive LWR design reviews. The application of these chemical forms in each evolutionary and passive LWR design review will be discussed in the staff's forthcoming safety evaluation reports.

ISSUE 3: **Equipment Survivability for Design Features Needed for Severe Accident Mitigation and Containment Integrity (for System 80+, AP600, and SBWR).**

As the staff concluded in SECY-90-016, the equipment and features provided for preventing and mitigating severe reactor accidents need not be subject to the rigorous 10 CFR 50.49 environmental qualification requirements. However, mitigation features and sampling systems for severe reactor accidents must be designed to provide a reasonable level of confidence that they will survive the severe accident environment for which they are intended and be able to operate over the timespan for which they are needed.

For the purpose of radiological assessments, the staff proposes to define the DBA radiation environment as that resulting from fission-product releases from coolant activity release, gap release, and in-vessel release. The staff is also defining the severe reactor accident radiation environment as that resulting from the above fission-product releases plus ex-vessel release and late-in-vessel release that are associated with vessel failure and core-concrete interaction.

If safety-related equipment is relied on to cope with severe accident situations, there should also be a reasonable level of confidence that this equipment will survive the severe accident environment, including radiation, for the period that it is needed to perform its intended function. The staff will evaluate the evolutionary and passive LWR vendors' identification of such equipment and features, their intended functions, location, and arrangement, shielding, and the period during which they are needed to perform on a case-by-case basis.

For issues related to severe accidents, such as the survivability of equipment needed for severe accident mitigation and sampling, the staff believes that the ex-vessel and late in-vessel releases can represent an important part of the source terms. However, for those evolutionary and passive designs that include reliable ex-vessel flooding mechanisms for the core debris, the ex-vessel contribution may be significantly reduced by aerosol scrubbing.

*The staff will address this part of Issue 3 by requiring that equipment and features needed for severe accident prevention, mitigation, and sampling be designed to provide a reasonable level of confidence that they will operate in a severe accident environment. This environment would include the ex-vessel release, with proper credit for features to mitigate that release, and the late in-vessel release (for System 80+, AP600, and SBWR).*

ISSUE 4: Radioactive Iodine Deposition on BWR Main Steamlines and Condensers (for SBWR).

The main steamlines in BWR plants, including the ABWR and SBWR, 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 of the valves will leak. Operating experience has indicated that the leaktightness of MSIVs has occasionally degraded and the leakage limits in the technical specifications have not always been maintained.

When calculating offsite consequences of potential accidents, the staff conservatively assumed that radioactive material passing through the MSIV at its technical specifications leakage limit is released directly into the environment. No credit was given for the integrity and leaktightness of the main steam piping and condenser to provide holdup and plateout of fission products.

Because of recurring problems with excessive leakage of MSIVs, the staff recommended in Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants," installation of a supplemental leakage control system (LCS) to ensure that the isolation function of the MSIVs is in accord with the specified limits. Most of the operating BWRs have an LCS.

In response to the MSIV leakage concerns, the BWR Owners Group commissioned a program of studies to determine the causes of high leak rates and the means to eliminate them. The results of these studies were submitted to the staff in General Electric (GE) proprietary reports, NEDO-31643P (November 1988) and NEDO-31858P (February 1991), both entitled "Increasing Main Steam Isolation Valve Leakage Rate Limits and Elimination of Leakage Control Systems."

EPRI and GE referenced these reports in the requirements documents and the ABWR and SBWR standard safety analysis reports (SSARs), respectively, as the basis for not providing an LCS and for requesting a substantially higher MSIV total leak rate [1.1 x $10^{-3}$ $m^3$/sec (140 $ft^3$/hr) rather than 3.5 x $10^{-4}$ $m^3$/sec (45 $ft^3$/hr)] than the typically allowed limit for operating BWRs. In addition, GE claimed in the ABWR and SBWR SSARs, that elemental and particulate iodine will deposit on the main steamlines (MSLs) and condenser, thus mitigating the radiological consequences of an accident. The basis for GE's claim was contained in a model developed for this purpose and documented in the referenced NEDO reports.

Following a LOCA, three potential release pathways exist for main steam leakage through the MSIVs:

(1) main steam drainlines to the condenser with delayed release to the environment through the low-pressure turbine seals

(2) turbine bypass lines to the condenser with delayed release to the environment through the low-pressure turbine seals

(3) main steamline turbine stop and control valves through the high-pressure turbine seals to the environment bypassing the condenser

The consequences of leakage from pathways 1 and 2 will be essentially the same because the condenser can be used to process MSIV leakage. The iodine removal efficiency of the condenser will vary depending on the inlet location of the bypass or drainline piping, but in either case, iodine will be removed. For pathway 3, MSIV leakage through the closed turbine stop and control valves will not be processed via the condenser. For this case, the high-pressure turbine (having a large internal surface area associated with the turbine blades) will remove iodine.

The staff believes that as long as either the turbine bypass or drainline leakage pathway is available, MSIV leakage through the closed turbine stop and control valves (pathway 3) will be negligible. Essentially all of the releases will be through the main condenser because there will be no differential pressure in the MSL downstream of the MSIVs following the closure of the valves.

Furthermore, MSIV leakage through pathway 3, if any, will have been subjected to the same iodine-removal processes in the MSLs (up to turbine stop valves) as those in the other pathways. The leakage will be further subjected to iodine removal by deposition in the high-pressure turbine internal surfaces. Removal by the main condenser is not applicable in pathway 3.

Basic principles of chemistry and physics predict that gaseous iodine and airborne iodine particulate material will deposit on surfaces. Several laboratory and in-plant studies have demonstrated that gaseous iodine will be deposited by chemical adsorption, and particulate iodine will be deposited through a combination of sedimentation, molecular diffusion, turbulent diffusion, and impaction. The transport of gaseous iodine in elemental and particulate forms also has been studied for many years, and several groups have proposed different models to describe the observed phenomena.

The staff has developed a model that treats the MSIV leakage pathway as a sequence of small segments for which instantaneous and homogeneous mixing is conservatively assumed; the mixing computed for each segment is passed along as input to the next segment. The number of segments depends on the parameters of the line and flow rate and can be as many as 100,000 for a long, large-diameter pipe and a low flow rate. Each line segment is divided into five compartments that represent the concentrations of the three airborne iodine species, the surface that contains iodine available for resuspension, and surface iodine that has reacted and is fixed on the surface.

The staff's model considers three iodine species: elemental, particulate, and organic. A fourth species, hypoiodous acid, was considered for the purpose of the staff's model to be a form of elemental iodine. All iodine in the segment undergoes radioactive decay. The resulting concentration from each segment of the deposition compartment serves as the input to the next segment. The

technical references indicate that particulate and elemental iodine would be expected to deposit on surfaces with rates of deposition varying with temperature, pressure, gas composition, surface material, and particulate size. Therefore, the staff believes that an appropriate credit for the removal of iodine in the MSLs and main condensers should be provided in the radiological consequence assessment following a design-basis accident.

In the review of the ABWR design, the staff accepted GE's proposed elimination of the LCS and allowed a higher MSIV leakage rate providing an appropriate credit for the removal of iodine in the MSLs and condenser. The staff gave a detailed technical evaluation in the ABWR draft final safety evaluation report (DFSER). Even though the staff used the TID-14844 source terms (mostly elemental iodine) for the ABWR design, as proposed by GE, the staff's model is also applicable for the NUREG-1465 source terms (mostly iodine in particulate forms as aerosol).

The GE model, as well as the one used by the staff, is based on time-dependent temperature adsorption phenomena with instantaneous and perfect mixing in a given volume. Both models use the same MSIV leakage pathways. They differ, however, in the treatment of buildup of iodine in the MSLs and condenser. GE assumed steady-state iodine in equilibrium in a large volume, while the staff assumed transient buildup of iodine in a finite number of small volumes. The staff does not consider these differences to be significant and finds that the resulting radiological consequences (offsite doses) are in good agreement. It therefore finds the GE model to be acceptable.

Sections III(c) and VI of Appendix A to 10 CFR Part 100 require 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 guidelines of Part 100 be designed to remain functional during and after a safe shutdown earthquake (SSE). Thus, the MSL, portions of its associated piping, and the main condenser are required to remain functional if credit is taken for deposition of iodine and if the SSE occurs. Consequently, the staff's past practice has been to classify these components as safety related and seismic Category I. In addition, 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.

In Section II.E, "Classification of Main Steamlines in Boiling Water Reactors," of SECY-93-087, the staff stated that the classification of the MSL 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 MSL from the outmost isolation valve up to, but not including, the turbine stop valve are equivalent to the staff guidelines in Appendix A to SRP Section 3.2.2, "System Quality Group Classification," and Regulatory Guide 1.29, "Seismic Design Classification."

In SECY-93-087 (Item II.E), the staff proposed for Commission approval, among other things, that (1) neither the main steam drain and bypass lines from the first valve up to the condenser inlet nor the piping between the turbine stop valve and the turbine inlet be classified as safety related or as seismic

Category 1; rather, these lines should be analyzed using a dynamic seismic analysis to demonstrate structural integrity under SSE loading conditions; and (2) the turbine stop, control, and bypass valves and the MSLs from the turbine control valves to the turbine meet all of the quality group and quality assurance guidelines specified in Appendix A to SRP Section 3.2.2; and (3) seismic analyses be performed to ensure that the condenser anchorages and the piping inlet nozzle to the condenser are capable of maintaining their structural integrity during and after the SSE. The staff concluded in SECY-93-087 that the above-described approach provides reasonable assurance that the main steam piping from the outmost isolation valve up to the turbine stop valve, the main steam drain and bypass lines up to the condenser, and the main condenser will retain their pressure and structural integrity during and following SSE. The Commission approved the above staff proposals in its staff requirements memorandum of July 21, 1993.

In its review of the ABWR design, the staff concluded in the FSER that the ABWR design provides reasonable assurance that the main steam piping from the outmost isolation valve up to the turbine stop valve, the MSIV leakage pathway (i.e., the drainline or bypass line) up to the condenser, and the main condenser will remain structurally intact and leaktight, so that they can act as a holdup volume for fission products during and following an SSE.

The staff also determined that the ABWR design meets the requirements of 10 CFR Part 100 because the structures, systems, and components described above are designed to remain functional during and following an SSE.

For the purpose of providing a credit for iodine holdup and plateout, the staff's model requires that the main steam piping (including its associated piping to the condenser) and the condenser remain structurally intact following an SSE, so they can act as a holdup volume for fission products. By the term "structurally intact," the staff assumes the steamline will retain sufficient structural integrity to transport the MSIV bypass leakage at its relatively low flow rate throughout the steamlines and condenser. The staff considers, in its radiological consequence assessment, that the condenser is open to the atmosphere via leakage through the low-pressure turbine seals. Thus, it is only necessary to ensure that gross structural failure of the condenser will not occur.

***The staff has addressed Issue 4 by accepting the passive BWR plant designs without an LCS and allowing an appropriate credit for iodine removal in the MSL and condenser following a design-basis accident (for SBWR).***

The robustness of the leak pathways and the amounts of iodine removed are plant design specific and, therefore, will be determined on a case-by-case basis.

**ISSUE 5: Fission-Product Holdup in the Secondary Containment (for SBWR).**

The structure or structures that completely surround the reactor primary containment and that are held at a negative pressure can be classified as a secondary building for the purpose of fission-product control and, therefore, for the mitigation of radiological consequences. The EPRI requirements

documents for evolutionary and passive plans designs require no holdup and retention of fission products in the PWR secondary building. Westinghouse has not claimed any credit for fission-product holdup in the AP600 auxiliary (secondary) building.

GE proposed a safety envelope design for the SBWR plant. This envelope is a reinforced concrete structure (secondary containment) within the reactor building that forms an envelope completely surrounding the primary containment (except the basement drywell head and drywell top slab). During normal operation, the safety envelope is maintained at a slightly negative pressure relative to the atmosphere by the reactor building heating, ventilation, and air conditioning system. The safety envelope is designed to be automatically isolated on detection of, among other things, a high radiation environment. Its design leakage rate is not to exceed 25 percent of the safety envelope free volume per day at a differential pressure of 6 mm (0.25 inch) of water.

GE stated in the SBWR standard safety analysis report that the safety envelope is designed to be capable of periodic testing to ensure its intended performance and that testing and inspection of the integrity of the safety envelope, including its leakage rate, will be performed in accordance with the specific plant technical specifications. To provide a credit for fission-product holdup, the staff will require periodic leak rate testing similar to that performed for the primary containment. For safety envelope air mixing, GE proposed 50 percent on the basis of one-dimensional modeling of the KEMA computer code using simplified compartment volumes. GE informed the staff that it will follow this simplified modeling with a three-dimensional modeling effort using a computational fluid dynamic computer code to better define the capability of the SBWR safety envelope to hold up radioactive material and mitigate any potential leakage. GE further informed the staff that this effort will be completed in 1994.

*The staff has addressed Issue 5 by allowing appropriate fission-product holdup (for decay) credit without requiring that a negative pressure be maintained in the SBWR secondary containment (building) if (1) the vendor specifies that the secondary containment leakage and mixing performance be consistent with the values used by the staff in its radiological assessment and (2) the COL combined license applicant incorporates the secondary containment leakage value specified by the vendor into the plant-specific technical specifications (for SBWR).*

**ISSUE 6: Fission-Product Release Timing (for System 80+, AP600, and SBWR)**

In SECY-90-309, "Impacts of Source Term Timing on NRC Regulatory Positions," the staff informed the Commission of its plans to change regulatory positions for both existing and future advanced reactor designs that result from more realistic treatment of fission-product release timing. In Regulatory Guides 1.3 and 1.4, the instantaneous release of fission products into the containment on receipt of a DBA signal from maximum full-power operation of the core is postulated and fission products released are assumed to be immediately available for leakage from the containment atmosphere to the environment.

The staff evaluated a number of items in SECY-90-309 to determine if more realistic treatment of fission-product releases would lead to less demanding regulatory requirements. The staff concluded that relaxation of requirements was justified for (1) the containment purge/vent isolation valve closure time, (2) the main steam isolation valve closure time, and (3) the emergency diesel generator start time. It also concluded that relaxing the timing requirements for each of these items would affect more than radiological consequences and, therefore, a plant-specific safety analysis would be required for each proposed relaxation.

In draft NUREG-1465, the staff indicated that fission-product gap activity release was estimated for a large-break LOCA to commence no earlier than 10 to 30 seconds for PWRs and 30 seconds for BWRs. It also indicated that fission-product early in-vessel release was estimated to commence no earlier than 0.5 and 1.0 hour for PWRs and BWRs, respectively. In contrast to the instantaneous releases that were postulated in Regulatory Guides 1.3 and 1.4, analyses of severe accident sequences have shown that fission-product releases can be generally categorized in terms of phenomenological phases associated with fuel failure, melting, and relocation despite differences in reactor design and accident sequence.

The single values for release timing and durations shown in draft NUREG-1465 have been chosen on the basis of simplicity, consistency, and conservatism. An accurate determination of the release timing and durations will depend not only on the reactor design but also on the selected accident sequences. The staff will review an evolutionary or passive reactor vendor's design and accident sequence as a basis for fission-product release timing to determine if its values are different from those provided in draft NUREG-1465. For its reviews of evolutionary and passive LWR designs, the staff will use the fission-product release timing values for the radiological consequence assessments of (1) DBAs, (2) the operation of control room habitability systems, (3) the operation of ESF filtration systems, (4) containment and MSIV closure, (5) containment purge/vent isolation valve closure, and (6) emergency diesel generator start time.

The staff will discuss the application of fission-product timing in each evolutionary and passive LWR design review in its forthcoming safety evaluation reports. In particular, the calculated onset of fission-product release may vary substantially depending on codes and assumptions. The staff proposes establishing limits on these timing ranges, so that a degree of conservatism will be ensured.

As a general guideline, the staff proposes to start the coolant activity release at the beginning of the accident, the gap activity release no later than 10 minutes into the accident (assuming approval of leak-before-break methodology by the staff), and the early in-vessel release no later than 30 and 60 minutes into the accident for PWRs and BWR, respectively. If applicable, the ex-vessel release should start at the time of postulated vessel failure. The late in-vessel release will be evaluated on a case-by-case basis.

Issue 7: Aerosol Deposition in Primary Containment (for System 80+, AP600, and SBWR)

The principal means of removing the airborne fission products from the containment atmosphere in the LWR design traditionally included use of active containment atmosphere cleanup systems such as containment spray, ESF filtration, and pressure-suppression pool scrubbing. In the TID-14844 source terms, the staff assumed that 50 percent of radioactive iodine (91 percent in the form of elemental iodine) released into the containment was immediately removed by plateout (diffusion mechanism). No fission products in aerosol form were assumed to be present. In the passive LWRs, an active containment atmosphere cleanup system is not provided. Reliance is placed solely on such natural aerosol removal processes in the containment as holdup (for decay), sedimentation (for settling), diffusion (for plateout), and leakage (for depletion).

In response to the staff's request for additional information, EPRI stated that because a containment spray system is not used for the passive plant, the means by which aerosol is removed will be dominated by such natural removal effects as diffusion and sedimentation.

The most complete mechanistic treatment of aerosol behavior in the reactor containment is found in CONTAIN, a computer code developed at Sandia National Laboratories under NRC sponsorship for the analysis of containment aerosol behavior under severe accident conditions. The other computer code, NAUA, which is very similar to the aerosol portion of CONTAIN, was developed at Kernforschungszentrum, Karlsruhe, Federal Republic of Germany, and was used for aerosol treatment in the NRC Source Term Code Package. There are a number of other aerosol behavior computer codes, but these two codes are the most widely used and accepted throughout the international nuclear safety community. Either code is acceptable to the staff for the evaluation of aerosol fission-product behavior in the primary containment following a severe reactor accident. However, most of these computer codes use lumped parameter methods, which assume that the aerosols as well mixed in the containment. When the condition of mixing cannot be established, the calculated results using the above computer codes may not be valid.

The staff will consider two natural processes for removing aerosols from the containment atmosphere over the entire period of an accident (30 days): (1) sedimentation mechanism of gravitational settling, including aerosol agglomeration, and (2) diffusion mechanisms of diffusiophoresis and thermophoresis. Diffusiophoresis is the diffusion of particles into interior heat sink surfaces, such as the containment walls on which steam is condensing. Thermophoresis is a similar effect due to a temperature gradient near the surface. In a paper on the source term for passive plants supplemented by a draft report titled "Passive ALWR Containment Natural Aerosol Removal" (April 1993), EPRI has provided radioactive aerosol removal rates based on sedimentation and diffusion of aerosols in the containment atmosphere using the EPRI version of the NAUA code (NAUAHYGROS). Westinghouse and GE also provided similar removal rates in the passive LWR standard safety analysis reports for AP600 and SBWR, respectively. GE stated that it will perform a specific aerosol removal analysis using the NAUA code (or a similar

computer code) for the SBWR containment design to justify the aerosol removal rates used in the SBWR standard safety analysis report.

The staff agrees with EPRI and the passive light water vendors on the physical processes associated with natural aerosol removal that could be taken into account in establishing the airborne fission-product concentrations in the containment atmosphere. Therefore, the staff accepts the aerosol removal mechanisms proposed by EPRI in the requirements documents for evolutionary and passive plant designs. However, the containment aerosol removal rates are plant design specific and will vary, depending on, but not limited to, the containment geometry, containment size, surface area, steam quality, and containment cooling mechanisms. The staff is reviewing the AP600 and SBWR passive reactor designs. In doing so, it will use available literature, expert opinions (e.g., NUREG-1150, NRC contractors, Organization for Economic Cooperation and Development reports), and engineering judgments. In particular, the staff will carefully review the selection of initial size distribution, particle density, applied agglomeration model, steam condensation on particles, and the effect of fission product solubility.

The removal mechanisms, as applied to the new source terms, will be scrutinized on the basis of the scientific soundness of the physical assumptions and the consistency with conservative practice of DBA calculations. The staff will accept any physically based analysis that conservatively estimates the amount of airborne fission products for the offsite dose calculations, and conservatively estimates the amount of settled fission products for environmental qualification calculations. The specific aerosol removal rates evaluated and used by the staff in the evolutionary and passive LWR radiological consequence assessments, complete with their technical justification on air-mixing efficiency in the containment, will be provided in the staff's forthcoming safety evaluation reports.

ISSUE 8: Aerosol Removal by BWR Suppression Pool (for ABWR and SBWR)

The BWR suppression pools are designed primarily as containment pressure and temperature suppression mechanisms for reactor pressure vessel blowdown. However, they can also serve as a medium for scrubbing radioactive fission products except noble gases and iodine in organic forms. The scrubbing (attenuation) of fission products in suppression pools is usually expressed as a "decontamination factor" (DF), which is defined as the ratio of the radioactive material injected into the pool divided by the airborne radioactive material that leaves the surface of the pool water.

Regulatory Guide 1.3, issued in 1974, does not recommend credit for fission-product scrubbing by BWR suppression pools. However, the Reactor Safety Study (WASH-1400), issued in 1975, assumed a DF of 100 for subcooled suppression pools and 1.0 for steam-saturated pools. Since 1975, the staff and EPRI have developed detailed models for the analysis of radioactive aerosol scrubbing by the suppression pool. Accordingly, in 1988, the staff issued revised SRP Section 6.5.5, "Pressure Suppression Pool as a Fission Product Cleanup System," stating that ignoring the large volume of research data supporting pool credit would be an undue degree of conservatism. The revised SRP section allowed a maximum DF of 10 for particulate and elemental iodines for Mark II

and Mark III containments and a DF of 5 for a Mark I containment. These DF values were based on the TID-14844 report, which stipulates an instantaneous release of fission products from the reactor coolant system to the containment atmosphere. In the revised SRP section, this was further subjected to the fraction of the drywell atmosphere bypassing the suppression pool by leaking through drywell penetrations to obtain the overall DF.

GE utilized the TID-14844 source terms for its ABWR design. The staff conservatively assumed in its radiological consequence assessment a DF of 2 by the ABWR suppression pool (equivalent to suppression pool steam bypass of 50 percent) for airborne radioactive iodine in elemental and particulate forms. The staff has recently accepted the drywell leakage values [0.005 $m^2$ (0.05 $ft^2$) of effective leakage area pathway] proposed by GE for the ABWR. The ABWR drywell and wetwell containment sprays in the drywell or wetwell, or both, also would reduce the effect of suppression pool bypass leakage on containment performance. Therefore, a DF of 2 used in the staff's analysis is conservative. For the ABWR design, the staff will review the plant-specific technical specifications, which require periodic inspections to confirm suppression pool depth, and surveillance tests to confirm drywell leaktightness on a case-by-case basis.

ISSUE 9: Fission-Product Removal by Containment Spray (For AP600)

General Design Criteria (GDC) 41, 42, and 43 of Appendix A to 10 CFR Part 50 require systems to control fission products to reduce the concentration that may be released to the environment. The containment spray system reduces containment pressure and temperature and removes airborne radioactive fission products in the containment atmosphere following a LOCA.

The EPRI requirements document for evolutionary plant designs requires a containment spray system. ABB-CE has provided an acceptable spray system for the System 80+ design. For its ABWR design, GE provided a safety-related spray system in the containment to reduce pressure and temperature following a LOCA, but has not claimed nor has the staff provided, any credit for the removal of airborne fission products for the purpose of radiological consequence assessments.

The EPRI requirements document for passive plant designs requires neither an active containment atmosphere cleanup system nor a containment spray system. This document states that fission-product control should be accomplished passively by reliance on natural deposition (plateout) and holdup (decay) and slower fission-product release timing. The Westinghouse AP600 design relies on plateout and holdup in the primary containment. In the SBWR design, GE relies on plateout in the reactor primary containment and holdup in the reactor primary containment and in the safety envelope. The safety envelope is a reinforced-concrete structure within the reactor building that forms an envelope completely surrounding the containment. It is designed to be periodically tested for leaktightness. The SBWR design includes a non-safety-grade spray system for both the upper drywell and suppression chamber. However, GE has not claimed, nor has the staff provided, fission-product removal credit for the non-safety-grade spray system in the SBWR design. The AP600 design does not provide a non-safety- grade containment spray.

The passive LWR vendors have submitted radiological consequence assessments to demonstrate that without containment spray, the passive LWR designs can meet the dose reference values of 10 CFR Part 100 and the control room operator dose limits in GDC 19 using the fission-product release timing based on the specific designs and selected accident sequences and using their best estimate of fission-product removal efficiencies. Unlike the current generation of operating PWRs, the Westinghouse AP600 design does not include containment sprays. The staff will review the AP600 radiological consequence assessment to determine if the dose reference values in 10 CFR Part 100 and the control room operator dose criteria of GDC 19 can be met without a containment spray system. That determination will involve evaluation of aerosol removal rates by natural deposition, containment leak rates, and assurance of pH control.

ISSUE 10: Radioactive Aerosol and Iodine Removal by Engineered Safety Features (ESF) Atmosphere Cleanup System (for AP600 and SBWR)

Containment atmosphere cleanup systems are to be provided as necessary to reduce the amount of radioactive material released to the environment following a postulated design-basis accident in accordance with GDC 41, 42, and 43. These GDC also require that these systems be designed to permit appropriate periodic inspection and testing to ensure their integrity, capability, and operability.

The EPRI requirements document for passive plant designs does not require an ESF atmosphere cleanup system, and the passive LWR vendors have not provided these systems in the AP600 and SBWR designs. The System 80+ design includes an ESF atmosphere cleanup system with charcoal adsorbers for the anulus building ventilation system. However, ABB-CE did not claim any fission-product removal credit for charcoal adsorbers; therefore, they will not be required to be in the CE System 80+ technical specifications. The GE ABWR design includes an ESF atmosphere cleanup system with HEPA filters and charcoal adsorbers for the reactor building (standby gas treatment system) and control room habitability system. Since the chemical forms of iodine will be predominantly in particulate form, HEPA filters rather than charcoal adsorbers will play a major role in removing airborne fission products following an accident.

The passive LWR vendors have submitted the radiological consequence assessments to demonstrate that without an ESF atmosphere cleanup system, the passive LWR designs can meet the dose reference values of 10 CFR Part 100 and the control room operator dose limits in GDC 19 using the fission-product release timing based on its specific passive LWR design and selected accident sequences and using their best-estimate fission-product removal efficiencies. The staff will review these assessments and provide its conclusions in the safety evaluation reports.

Therefore, the staff will not require an ESF atmosphere cleanup system, provided the passive LWR vendor demonstrates that the passive LWR design, including adequate pH control can meet the dose reference values in 10 CFR Part 100 and the control room operator dose limits in GDC 19.

ISSUE 11: Atmospheric Diffusion Model for Control Room Habitability Assessment (for System 80+, AP600, and SBWR)

The staff has developed with contractor assistance a new model for determining atmospheric relative concentrations ($\chi/Q$) in building wakes to be used for control room habitability assessment. The new model is applicable for either TID-14844 or revised accident source terms and may be used by the evolutionary and passive LWR vendors and existing licensees. The staff will use this model in its review of evolutionary and passive LWR designs using the given building layouts and standard plant site design parameters.

The current guidelines for determining $\chi/Q$ are given in Regulatory Guides 1.3, 1.4, and 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and in SRP Section 6.4, "Control Room Habitability Systems." The new model, in general, is more realistic in determining the atmospheric dispersion factors for assessing concentrations of radioactive material at the control room emergency air intake except for the reactor sites with a high average wind speed (e.g., sites around the Great Lakes in the United States). This model is not proposed for backfitting on existing plants. However, because it reflects the current understanding of building wake behavior, the staff plans to make this new model available to existing licensees who may voluntarily propose to use it in future license amendments. The staff will use this new model for reviewing future reactor designs and their siting.

The new model differs from the current model in that it assumes the wind will transport radioactive materials directly from the release point to the control room air intake only if the wind direction is within ± 30 degrees of the direction needed to transport radioactive material directly to the intake. It also determines 2-hour and 4-hour average concentrations from hourly centerline $\chi/Q$ values. It will use hourly, sector-average $\chi/Q$ values to calculate average concentrations for periods of 24, 48, 96, 168, 360, and 720 hours. All averages will be computed as moving (overlapping) averages of the appropriate hourly $\chi/Q$ values.

The staff will use the new model in determining the atmospheric dispersion factors for assessing concentrations of radioactive material at the control room emergency air intake following a design-basis accident for evolutionary and passive LWR design reviews.

ISSUE 12: Failure of Heat Exchanger Tubes in the Passive Containment Cooling System (for SBWR)

The SBWR passive containment cooling system (PCCS) will remove the core decay heat rejected to the containment after a LOCA. It consists of three independent closed loops that are extensions of the containment. Each loop contains a heat exchanger that condenses steam on the tube side and transfers heat to the water in a large pool located outside the containment. The PCCS loops are operated at a low pressure and are initially driven by the pressure difference created between the containment drywell and suppression pool and then by gravity drainage of steam condensed in the tubes. Three PCCS loops are an extension of the containment and do not have isolation valves. The

potential for containment bypass in this manner only applies to the SBWR design.

Should the PCCS heat exchanger tubes fail, the PCCS will provide a bypass pathway for the SBWR containment, releasing radioactive fission products from the containment atmosphere to the reactor building through the passive containment cooling (PCC) pool water. The staff expects the radioactive aerosol scrubbing by the PCC pool water to be similar to that by the BWR pressure-suppression pools.

The staff has not completed its review of the SBWR PCCS design, nor has it determined the important characteristics (such as carrier fluid conditions, failure configuration, pathway evaluation, and pool thermal and hydraulic characteristics) affecting the radioactive aerosol removal (source term behavior) by the PCC pool water. The staff's evaluation of this potential containment bypass with its source term behavior will be given in the safety evaluation report for the SBWR. The staff considers this failure event a new DBA for the passive LWR design. The discussion in this section provides information relative to staff commitments identified in SECY-93-087.

# DRAFT

ENCLOSURE 2

## RELEASE FRACTIONS FOR PROPOSED REACTOR ACCIDENT SOURCE TERMS

Table 1 BWR Releases Into Containment*

| | Gap Release | Early In-Vessel |
|---|---|---|
| Duration (Hours) | 1.0 | 1.5 |
| Noble Gases | 0.05 | 0.95 |
| Iodine | 0.05 | 0.22 |
| Cesium | 0.05 | 0.15 |
| Tellurium | 0 | 0.11 |
| Strontium | 0 | 0.03 |
| Barium | 0 | 0.03 |
| Ruthenium | 0 | 0.007 |
| Cerium | 0 | 0.009 |
| Lanthanum | 0 | 0.002 |

*Values shown are fractions of core inventory.

Table 2 PWR Releases Into Containment*

| | Gap Release | Early In-Vessel |
|---|---|---|
| Duration (Hours) | 0.5 | 1.3 |
| Noble Gases | 0.05 | 0.95 |
| Iodine | 0.05 | 0.35 |
| Cesium | 0.05 | 0.25 |
| Tellurium | 0 | 0.15 |
| Strontium | 0 | 0.03 |
| Barium | 0 | 0.04 |
| Ruthenium | 0 | 0.008 |
| Cerium | 0 | 0.01 |
| Lanthanum | 0 | 0.002 |

*Values shown are fractions of core inventory.

COMPARISON OF GAP AND IN-VESSEL RELEASES
IN NUREG-1465 WITH TID-14844 SOURCE TERMS FOR PWRs

TABLE 3

| Nuclide | TID-14844 | NUREG-1465 CE SYSTEM 80+ | AP600 |
|---|---|---|---|
| Noble Gases | 1.0 | 1.0 | 1.0 |
| Iodine | 0.5 | 0.4 | 0.30 |
| Cesium | <0.01 | 0.3 | 0.08 |
| Tellurium | <0.01 | 0.15 | 0.08 |
| Strontium | <0.01 | 0.03 | 0.004 |
| Barium | <0.01 | 0.04 | 0.004 |
| Ruthenium | <0.01 | 0.008 | 0.004 |
| Cerium | <0.01 | 0.01 | 0.009 |
| Lanthanum | <0.01 | 0.002 | 0.002 |

COMPARISON OF GAP AND IN-VESSEL RELEASES
IN NUREG-1465 TID-14844 SOURCE TERMS FOR BWRs

TABLE 4

| Nuclide | TID-14844 | NUREG-1465 | SBWR |
|---|---|---|---|
| Noble Gases | 1.0 | 1.0 | 1.0 |
| Iodine | 0.5 | 0.27 | 0.30 |
| Cesium | <0.01 | 0.20 | 0.23 |
| Tellurium | <0.01 | 0.11 | 0.06 |
| Strontium | <0.01 | 0.03 | 0.003 |
| Barium | <0.01 | 0.03 | 0.003 |
| Ruthenium | <0.01 | 0.007 | 0.0003 |
| Cerium | <0.01 | 0.009 | <0.0003 |
| Lanthanum | <0.01 | 0.002 | <0.0003 |

# REPORT BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS ON DRAFT COMMISSION PAPER, "SOURCE TERM RELATED TECHNICAL AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND PASSIVE LIGHT-WATER-REACTOR DESIGNS" MARCH 15, 1994

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

March 15, 1994

The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Selin:

SUBJECT: DRAFT COMMISSION PAPER ON SOURCE TERM RELATED TECHNICAL AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND PASSIVE LIGHT WATER REACTOR DESIGNS

During the 406th and 407th meetings of the Advisory Committee on Reactor Safeguards, February 10-11 and March 10-12, 1994, respectively, we discussed the draft Commission paper on source term related technical and licensing issues pertaining to evolutionary and passive light water reactor (LWR) designs. During these meetings, we had the benefit of discussions with representatives of the NRC staff and industry. We also had the benefit of the documents referenced.

Separate source terms are provided for BWRs and PWRs. The source terms consist of the fraction of the equilibrium core inventory of fission products released into containment, the timing of this release, and the chemical form of the fission product iodine. In the past, such source terms have been specified in Regulatory Guides 1.3 and 1.4 to provide guidance on appropriate values to use in the site suitability analyses that are required by 10 CFR Part 100, and in conjunction with the other design basis accidents (DBAs) in Chapter 15 of the Standard Review Plan. The DBA source terms should not be confused with the plant and sequence specific source terms that are mechanistically derived and used in PRAs and other severe accident analyses. The specifications that are presently in Regulatory Guides 1.3 and 1.4 consist of 100 percent of the noble gases and 25 percent of the iodine (91 percent as elemental iodine, 5 percent as particulate iodine, and 4 percent as organic iodine). For site suitability analyses, these specifications have been used along with a thermal hydraulic specification. These analyses require that a peak containment pressure be calculated for a double-ended break of the largest primary system piping and be applied for 24 hours after which it is to be reduced to half that value.

The 10 CFR Part 100 specifications of the source term have always been viewed as being somewhat arbitrary, but conservative. The proposed revised source terms are intended to remove some of the arbitrariness of the present values and to make them more realistic. As part of the overall process of decoupling site suitability decisions from reactor design, the revised source term and the dose criteria provisions are to be removed from 10 CFR Part 100 and put into 10 CFR Part 50 where they would apply only to design features. The revised source terms are based on values developed in NUREG-1150 for the "in-vessel" release phase associated with severe accidents.

In the draft Commission paper, the staff describes the proposed revised source terms and proposed uses for reviews and assessments of evolutionary and passive LWR designs. The paper discusses positions taken by the staff on source term issues for evolutionary and passive LWR designs (identified in SECY-90-016 and SECY-93-087). The staff believes these positions will provide a basis for closing these issues with respect to design certification reviews and the EPRI Utility Requirements Documents.

We generally agree with the positions taken by the staff on the issues and agree with the principle that the source terms for DBAs should be made more realistic. Realistic source terms should result in more appropriate designs (e.g., engineered safety features, source term mitigation features, sampling and measurement devices, and containment integrity). We believe the changes can lead to increased coherence in the associated regulations and their application. As in all responses to the accumulation of new knowledge, such proposed changes in the regulations, whether toward enhancement or relaxation, or whether applied to existing plants or to future plants, should be assessed for their overall effect on risk. We also have the following concern about the revised source term specifications.

We think the realistic specification of the thermal hydraulics and production of nonradioactive aerosols associated with the DBAs is as important as the specification of the source term itself. These conditions can strongly influence the behavior of radioactive aerosols in containment. Additional consideration should be given to developing Commission guidance on the thermal hydraulic conditions and nonradioactive aerosol generation to be coupled with the source terms for the various DBAs.

We continue to recommend that the General Design Criteria for containment volume and strength for future ALWRs incorporate the spectrum of severe accident challenges described in our report of

May 17, 1991. The containment should represent a defense-in-depth feature that is not limited to design basis accidents.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman

References:

1. Memorandum dated January 6, 1994, from Dennis M. Crutchfield, NRC Office of Nuclear Reactor Regulation, for John T. Larkins, Executive Director, ACRS, Subject: ACRS Review of Commission Paper on Source Term-Related Technical and Licensing Issues Pertaining to Evolutionary and Passive Light-Water-Reactor Designs
2. Memorandum dated February 10, 1994, from James M. Taylor, NRC Executive Director for Operations, for the Commissioners, Subject: Draft Commission Paper, "Source Term Related Technical and Licensing Issues Pertaining to Evolutionary and Passive Light-Water-Reactor Designs"
3. SECY-93-087, Memorandum dated April 2, 1993, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs
4. SECY-90-016, Memorandum 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
5. NUREG-1150, Volumes 1 and 2, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power plants," December 1990
6. Report dated May 17, 1991, from David A. Ward, ACRS Chairman, to Kenneth M. Carr, NRC Chairman, Subject: Proposed Criteria to Accommodate Severe Accidents in Containment Design

# STAFF RESPONSE TO THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS ON DRAFT COMMISSION PAPER, "SOURCE TERM RELATED TECHNICAL AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND PASSIVE LIGHT-WATER-REACTOR DESIGNS" MAY 20, 1994

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 20, 1994

Dr. Thomas S. Kress, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Dr. Kress:

SUBJECT: DRAFT COMMISSION PAPER ON SOURCE-TERM RELATED TECHNICAL AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND PASSIVE LIGHT WATER REACTOR DESIGNS

I am responding to the Advisory Committee on Reactor Safeguards' (ACRS') letter to the Chairman dated March 15, 1994, in which you commented on the staff's draft Commission paper pertaining to the application of the revised accident source terms to evolutionary and passive light water reactor (LWR) designs.

In the draft Commission paper, the staff discussed 12 source term-related technical and licensing issues pertaining to either evolutionary LWRs or passive LWRs or both. The staff had previously identified all of these issues in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," and notes the ACRS' general agreement with its positions on the issues. Although the ACRS' letter did not specifically address any of the 12 issues discussed in the draft Commission paper, you did offer two general comments associated with the use of the new physically based source term in ALWRs. In the enclosure, the staff responds in detail to each of your comments.

As stated in the enclosure, the staff recognizes the importance of thermal hydraulics and the production of nonradioactive aerosols in determining the behavior of radioactive aerosols in containment. The staff is working to resolve these details during its current review of the applications for final design approval/design certification.

With regard to the ACRS' comment concerning the design criteria for containment volume and strength, the staff recognizes the advantages which would be achieved by putting severe accidents on an equal regulatory footing with design basis accidents. This approach was included as one of the options in the Advanced Notice of Proposed Rulemaking, "Acceptability of Plant Performance for Severe Accidents; Scope of Consideration in Safety Regulations." Currently, the staff is following the Commission's guidance in this matter.

The staff believes that its positions, as described in the draft Commission paper, are an important first step in applying the new source term to future ALWRs.

Sincerely,

Original signed by
James M. Taylor

James M. Taylor
Executive Director
for Operations

Enclosure:
As stated

cc w/enclosure:
Chairman Selin
Commissioner Rogers
Commissioner Remick
Commissioner de Planque
SECY

## RESPONSE TO ACRS COMMENTS ON THE DRAFT COMMISSION PAPER ON SOURCE TERM-RELATED TECHNICAL AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND PASSIVE LIGHT WATER REACTOR DESIGNS

1. ACRS COMMENT: We think the realistic specification of the thermal hydraulics and production of nonradioactive aerosols associated with the DBAs [design-basis accidents] is as important as the specification of the source term itself. These conditions can strongly influence the behavior of radioactive aerosols in containment. Additional consideration should be given to developing Commission guidance on the thermal hydraulic conditions and nonradioactive aerosol generation to be coupled with the source terms for the various DBAs.

   STAFF RESPONSE: The staff agrees that thermal hydraulic conditions and production of nonradioactive aerosols can strongly influence the behavior of radioactive aerosols in the containment following a DBA. The thermal hydraulic conditions include the containment pressure, relative humidity, and steam condensation and heat removal rates to the containment structure. These thermal hydraulic parameters, as well as the amounts of nonradioactive aerosols produced, differ with the specific reactor accident sequences and with the accident mitigation features provided (e.g., isolation condensers and primary containment cooling system for the simplified boiling water reactor (SBWR) design, containment shell cooling for the AP600 design, and containment spray for the Combustion Engineering (CE) System 80+ design).

   Because the AP600 and SBWR designs have been submitted and are under review, the staff does not intend to develop and promulgate specific guidance relative to the impact of thermal hydraulic conditions and production of nonradioactive aerosols on the behavior of radioactive aerosols in containment at this time. It will, however, communicate on these matters with Westinghouse and General Electric (GE) via requests for additional information (RAIs) or similar licensing communication vehicles during the development of the draft safety evaluation reports. In the discussion that follows, the staff has summarized the current status of the three ALWR design reviews that are based on the revised accident source terms.

   GE proposed a group of reference accident sequences for the SBWR design that lead to core damage but terminate with the reactor pressure vessel still intact to determine the thermal hydraulic conditions in the containment following the DBA. GE, in turn, used these reference accident sequences to generate the thermal hydraulic conditions using an SBWR version of the MAAP code. The staff is reviewing the GE proposal and will either select its own representative group of accident sequences or, after appropriate review, use the GE-selected accident sequence for its independent determination of the thermal hydraulic conditions and nonradioactive aerosol generation. The staff will use

Enclosure

this information along with the following parameters for evaluating the fission-product behavior and determining the aerosol removal rates in the SBWR containment following a DBA:

- containment geometry
- aerosol characteristics
- aerosol removal by isolation condensers and primary containment cooling systems

Westinghouse has provided neither DBA sequences nor the thermal hydraulic conditions in the containment for the AP600 design. The staff is addressing this matter with Westinghouse by means of RAIs.

The CE System 80+ evolutionary plant design includes a safety-grade containment spray system; therefore, the thermal hydraulic conditions and the amounts of nonradioactive aerosol are less significant for determining the behavior of radioactive aerosol behavior and its removal rates in the containment. Nevertheless, in its evaluation of the System 80+ design, the staff used a steam condensation rate of 100 moles per second (typical value from NUREG-1150) and 350 kilograms (770 lbs) of nonradioactive aerosol (typical value from draft NUREG-1465).

2. ACRS COMMENT: We continue to recommend that the general design criteria (GDC) for containment volume and strength for future ALWRs incorporate the spectrum of severe accident challenges described in our report of May 17, 1991. The containment should represent a defense-in-depth feature that is not limited to design basis accidents.

STAFF RESPONSE: The staff included this approach as one of three options selected to be incorporated into the Advanced Notice of Proposed Rulemaking (ANPR) titled "Acceptability of Plant Performance for Severe Accidents; Scope of Consideration in Safety Regulations," which was published in the Federal Register on September 28, 1992.

In SECY-93-226, "Public Comments on 57 FR 44513 - Proposed Rule on ALWR Severe Accident Performance," the staff summarized and discussed the public comments on the ANPR and recommended a proposed approach for proceeding with the drafting of a generic rule. In its staff requirements memorandum dated September 14, 1993, the Commission approved the staff recommendation in SECY-93-226 to delay a decision on the need for generic rulemaking to address severe accidents at least until after the final safety evaluation reports are issued for the ABWR and the System 80+. Further, the Commission recommended that any such generic rulemaking follow completion of the revisions to 10 CFR Part 50 and Part 100. Therefore, the staff intends to follow the direction of the Commission in this matter and will continue further dialogue with the ACRS at an appropriate time in the future.

# ANNEX D

## DRAFT COMMISSION PAPER
## "EMERGENCY PLANNING (EP) UNDER 10 CFR PART 52"
## MAY 23, 1994

May 23, 1994

**DRAFT**

**FOR:** The Commissioners

**FROM:** James M. Taylor
Executive Director for Operations

**SUBJECT:** EMERGENCY PLANNING (EP) UNDER 10 CFR PART 52

**PURPOSE:**

To inform the Commission of the staff's views on how EP requirements will be addressed at each phase of nuclear power plant licensing under 10 CFR Part 52.

**SUMMARY:**

The Office of Nuclear Reactor Regulation (NRR) has a general concept for addressing EP in its reviews of applications for early site permits, design certifications, and combined licenses under 10 CFR Part 52. The Federal Emergency Management Agency (FEMA) will be a partner in these actions to review offsite aspects of EP for the issuance of early site permits and combined licenses.

Early Site Permits

The FEMA and NRC staffs have drafted joint criteria, in Supplement 2 to NUREG-0654/FEMA-REP-1, Revision 1, for both agencies to use in reviewing

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504-2907

applications for early site permits. On June 17, 1993, the NRC and FEMA signed a revised FEMA/NRC memorandum of understanding on radiological emergency planning and preparedness that describes, among other things, joint activities of the two agencies in processing EP aspects of an early site permit.

Standard Design Certifications

The NRR staff has identified inspections, tests, analyses, and acceptance criteria (ITAAC) pertinent to emergency response onsite facilities in applications for standard design certifications and is reviewing such applications. These reviews deal solely with plant features; therefore, FEMA is not involved.

Combined Licenses

With regard to EP in combined licenses, the principal issues involve the form and role of ITAAC and the treatment of preoperational EP exercises. An application for a combined license must include proposed inspections, tests, analyses, and acceptance criteria (ITAAC) including those applicable to EP [52.79(c)]. These ITAAC are to be necessary and sufficient to demonstrate compliance with the 16 emergency planning standards in 10 CFR 50.47(b), or any acceptable alternatives as provided for in 10 CFR 50.47(c). Appendix E to 10 CFR Part 50 also requires that a "full participation" emergency preparedness exercise be conducted within 2 years before the first reactor unit at a site is authorized to operate above 5 percent of rated power. That exercise will be included in the combined license ITAAC. In the course of construction, the NRC staff will ensure that ITAAC applicable to licensee EP are satisfied and FEMA will ensure that ITAAC applicable to offsite EP are satisfied. (The NRC and FEMA staffs will need to develop procedures for the respective ITAAC reviews they will do during plant construction.)

**DISCUSSION**:

Early Site Permits

Subpart A to 10 CFR Part 52 states requirements and procedures for issuing an early site permit (ESP) that approves a site for one or more nuclear power plants separate from the filing of an application for a construction permit or a combined construction and operating license.

The application for an ESP must identify physical characteristics unique to the proposed site, such as egress limitations from the area surrounding the site, that could pose a significant impediment to the development of emergency plans. The application may also propose major features of the emergency plans, such as the exact size of the emergency planning zones, that can be reviewed and approved by NRC in consultation with FEMA, or may propose complete and integrated emergency plans for the site for review and approval

by NRC in consultation with FEMA. The application must include a description of contacts and arrangements made with local, State, and Federal governmental agencies that have emergency planning responsibilities.

The Commission is to determine, after consultation with FEMA, whether the application shows that there is no significant impediment to the development of emergency plans, whether any major features of emergency plans that may be submitted in the application are acceptable, and whether any emergency plans that may be submitted by the applicant offer reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency.

The NRC and FEMA staffs have jointly prepared guidance for writing, reviewing, and approving emergency information and plans to be submitted with an ESP application. The NRC and FEMA plan to issue this guidance in Supplement 2 to NUREG-0654/FEMA-REP-1, Revision 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants." The proposed Supplement 2 to NUREG-0654, containing emergency planning guidance for ESP applications, will be announced in the Federal Register and issued for public comment before final publication. (See Enclosure 1 for a draft of this document.) The staffs also revised the FEMA/NRC memorandum of understanding (MOU) on radiological emergency planning and preparedness to describe joint activities of the agencies in the processing of an ESP. On February 16, 1993, the Commission approved these revisions. These were incorporated, along with other changes approved by the Commission on May 27, 1993, into a new MOU, signed by NRC's Executive Director for Operations and FEMA's Acting Associate Director, on June 17, 1993. FEMA published the new MOU in the Federal Register on September 14, 1993 (58 FR 47996).

The staff notes that an ESP application is to include an evaluation of alternative sites [10 CFR 52.17(a)(2)]. In accordance with previous Commission guidance (staff requirements memorandum of April 9, 1991), emergency planning considerations will be included in the alternative site review process.

Standard Design Certifications

Emergency planning is not specifically discussed in Subpart B of 10 CFR Part 52, which deals with standard design certifications. Nevertheless, an application for design certification must contain "demonstration of compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f)." That rule requires licensees to provide an onsite Technical Support Center (TSC) and an onsite Operational Support Center (OSC) for the support of an emergency response. A TSC and OSC are among those facilities required by NRC's emergency planning rule [10 CFR 50.47 (b)(8)].

An application for design certification must contain "proposed ITAAC which are necessary and sufficient to provide reasonable assurance that, if the tests, inspections, and analyses are performed and the acceptance criteria met, a plant which references the design is built and will operate in accordance with the design certification." Accordingly, applications for design certification are to include ITAAC for the TSC and OSC. Enclosure 2 contains examples of possible ITAAC for the TSC and OSC. In general, the acceptance criteria in these sample ITAAC are based upon applicable criteria in NUREG-0696, "Functional Criteria for Emergency Response Facilities." Both the ABWR and ABB-CE System 80+ standard plant design applications contain ITAAC similar to those shown in Enclosure 2.

Combined Licenses

Subpart C of 10 CFR Part 52 states the requirements and procedures for issuing combined licenses for nuclear power facilities. In accordance with 10 CFR 52.83, all requirements of 10 CFR Part 50 and its appendices, applicable to holders of construction permits and operating licenses, also apply to holders of Part 52 combined licenses. The requirements for EP in Part 50 are contained principally in 10 CFR 50.47 and Appendix E of 10 CFR Part 50. A combined license may incorporate, by reference, an early site permit (ESP). A combined license may also reference a certified standard design previously approved under Part 52. In this way, the Commission may approve aspects of emergency planning for a nuclear power reactor in an ESP and may approve plant features supporting emergency response, such as an onsite Technical Support Center and an onsite Operational Support Center, in a certified design.

Emergency Plans in Applications for Combined Licenses

An application for a combined license is to include a final safety analysis report containing, among other things, emergency plans that contain the items specified in 10 CFR Part 50, Appendix E [as referenced in 10 CFR 50.34(b)(6)(v), which is referenced in 10 CFR 52.79(b)]. The application is to include radiological emergency response plans of State and local governmental entities that are wholly or partially within the plume exposure pathway emergency planning zone, and plans of State governmental entities that are wholly or partially within the ingestion emergency planning zone for the nuclear power reactor [10 CFR 50.33(g) as referenced in 10 CFR 52.77]. If the application does not reference an ESP or if no emergency plans were approved in connection with the issuance of the ESP, 10 CFR 52.79(d) requires that the application contain any certifications obtained from local and State governmental agencies with emergency planning responsibilities that (1) the proposed emergency plans are practicable, (2) these agencies are committed to participating in any further development of the plans, including any required field demonstrations, and (3) these agencies are committed to executing their responsibilities under the plans in the event of an emergency. If these certifications cannot be obtained, the application must contain information, including a utility plan, sufficient to show that the proposed plans nonetheless provide reasonable

assurance that adequate protective measures can and will be taken in the event of a radiological emergency at the site [10 CFR 52.79 (d)(2)].

Review of Applications for Combined Licenses

A combined license contains the attributes of both a construction permit and an operating license. Therefore, as stated in 10 CFR 52.81, the NRC will review applications for combined licenses according to existing regulatory standards for construction permits and operating licenses for nuclear power plants. NRC and FEMA will review the emergency plans proposed in the license application against the requirements in 10 CFR 50.47 and Appendix E of 10 CFR Part 50. The NRC staff will conduct reviews in accordance with Section 13.3 of the Standard Review Plan (NUREG-0800).

The NRC may not issue an operating license for a nuclear power reactor without finding reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency [10 CFR 50.47(a)]. The NRC reviews FEMA findings and determinations as to whether State and local emergency plans are adequate and whether there is reasonable assurance that they can be implemented. The NRC also assesses the applicant's onsite emergency plans to determine whether they are adequate and whether there is reasonable assurance that they can be implemented. The procedures for interactions between NRC and FEMA in the licensing process are specified in the NRC/FEMA MOU on radiological emergency planning and preparedness.

Applications Lacking Certifications From State and Local Agencies

Applications that do not contain all the certifications from State and local governmental agencies required by 10 CFR 52.79(d) must contain a utility plan that will be evaluated in accordance with 10 CFR 50.47(c)(1)(iii). In 10 CFR 50.47(c)(1), the NRC describes how an applicant can obtain an operating license if State or local governments decline or fail to participate in offsite radiological planning and preparedness. The applicant may demonstrate that deficiencies in the plans are not significant, that adequate interim compensating actions have been or will be taken promptly, or that there are other compelling reasons to permit plant operations. Compensating actions typically involve a utility plan. In determining the adequacy of such a plan, the NRC will recognize that, in an actual emergency, State and local government officials will exercise their best efforts to protect the health and safety of the public.

ITAAC in a Combined License

Under 10 CFR 52.79(c), the application for a combined license must include proposed ITAAC. Where the application references a certified standard design, the ITAAC contained in the certified design must apply to those portions of the facility covered by the design certification. The NRC, in consultation with FEMA, will review and approve emergency planning ITAAC proposed by an applicant for a combined license. The Commission will identify the required

ITAAC in the combined license [10 CFR 52.97(b)]. In accordance with 10 CFR 52.99 and 52.103(c), after the Commission issues a combined license, and before the fuel is loaded, the NRC staff will ensure that the required inspections, tests, and analyses are performed and that the prescribed acceptance criteria are met. Holders of combined licenses are required to facilitate NRC inspection at the site in accordance with 10 CFR 50.70, and to maintain records and reports in accordance with 10 CFR 50.71. At appropriate intervals during construction, the NRC staff will publish in the Federal Register notices of the successful completion of inspections, tests, and analyses.

The NRC recognizes that, to issue a combined license, it will need to make findings, before the completion of plant features required for emergency response and before the emergency plans are complete for such a facility, as provided in PL 102-486, "Energy Policy Act of 1992." ITAAC facilitate this by enabling the NRC, in consultation with FEMA, to make a predictive regulatory finding of reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. This finding is only subject to satisfactorily completing ITAAC before fuel load. Without such a finding, the NRC may not issue a combined license [10 CFR 50.47(a)(1), as referenced in 10 CFR 52.97(a)]. The ITAAC apply to the onsite emergency plans of the licensee and to offsite emergency plans of State and local governments or offsite plans of the utility. Accordingly, they will be reviewed and approved by the NRC in consultation with FEMA. FEMA may also request help from other Federal agencies through the regional assistance committees it has established under 44 CFR Part 350.

The applicant is to propose ITAAC in the combined license application. The Commission will identify, in the combined license, the ITAAC necessary and sufficient, to demonstrate that the emergency plans meet the 16 emergency planning standards in 10 CFR 50.47(b) and the requirements of Appendix E of 10 CFR Part 50. The acceptance criteria for each license will be based upon NUREG-0654/FEMA-REP-1, Revision 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants." Enclosure 3 contains preliminary drafts of illustrative EP ITAAC, addressing 2 of the 16 planning standards. Substantial additional developmental work on these ITAAC will be necesary. After issuing a combined license, the NRC and FEMA staffs will ensure that the required inspections, tests, and analyses pertaining to "onsite" and "offsite" EP, respectively, are performed. (The NRC and FEMA staffs will need to develop procedures for the respective ITAAC reviews they will do during plant construction.) Before the facility is operated, the NRC staff, in consultation with FEMA, must find that the prescribed acceptance criteria are met.

Emergency Preparedness Exercise

For the first reactor at a site, Appendix E requires that a full-participation exercise be conducted within 2 years before the NRC issues an operating license for full power (authorizing operation above 5 percent of rated power)

[10 CFR Part 50, Appendix E, Section IV.F.1]. Since this EP exercise would be included in the combined license ITAAC, it would have to be satisfied, along with other ITAAC, before fuel loading.

Combined License Proceedings

Among other things, ITAAC will be subject to review in any combined license hearings under Section 189a of the Atomic Energy Act, as amended. Such hearings will normally be conducted as part of combined license proceedings, well in advance of plant construction.

Once a combined license is issued, the next opportunity for a hearing normally occurs about six months before fuel loading. As provided for in 10 CFR 52.103(a), the Commission shall publish a notice of intended operation, which shall provide that any person whose interest may be affected by operation of the plant may, within 60 days, request the Commission to hold a hearing on whether the facility as constructed complies, or on completion will comply, with the acceptance criteria of the license. Compliance with the acceptance criteria applicable to emergency plans could be raised as an issue in such a proceeding. The petitioner is required pursuant to 10 CFR 52.103(b) to show, prima facie, that (1) one or more of the acceptance criteria in the combined license have not been or will not be met; and (2) the specific operational consequences of nonconformance that would be contrary to providing reasonable assurance of adequate protection of the public health and safety.

As provided for in 10 CFR 52.103(c), if the Commission grants a request for a hearing, it shall determine, after considering petitioners' prima facie evidence and any answers thereto, whether there will be reasonable assurance of adequate protection of the public health and safety during a period of interim operation. If the Commission determines that there is such reasonable assurance, it may allow operation during an interim period under the combined license. To the maximum possible extent, the Commission shall render a decision on the hearing issues within 180 days or by the scheduled date for fuel loading, whichever is later [10 CFR 52.103(e)].

FEMA could be involved in hearings associated with a combined license. The MOU between FEMA and the NRC on radiological emergency planning and preparedness, specifies that FEMA will give the NRC findings and determinations concerning offsite planning and preparedness related to reactor license reviews. The MOU also states, "FEMA will make expert witnesses available before the Commission, the NRC Advisory Committee on Reactor Safeguards, NRC hearing boards and administrative law judges, for any court actions, and during any related discovery proceedings."

Apart from a hearing under 10 CFR 52.103(b), a petition to modify the terms and conditions of the combined license may be submitted under 10 CFR 52.103(e). However, as indicated in 10 CFR 52.103(e), such a petition will be processed as a request for action under 10 CFR 2.206.

Operation Under a Combined License

After the Commission finds that the required inspections, tests and analyses are performed and the prescribed acceptance criteria are met [10 CFR 52.99], the provisions of 10 CFR Part 50 and its appendices applicable to operating licenses will govern operation under a combined license [10 CFR 52.83].

**COORDINATION:**

OGC has reviewed this paper and has no legal objection. The staff has also briefed the ACRS on the subject material in this paper at its August 5, 1993, meeting.

James M. Taylor
Executive Director
for Operations

Enclosures:
1. Draft Supplement 2 to NUREG-0654
2. Example of Certified Standard Design Related ITAAC
3. Illustrative ITAAC on Emergency Planning

NUREG-0654
FEMA-REP-1
Rev. 1
Supp. 2

# CRITERIA FOR PREPARATION and EVALUATION of RADIOLOGICAL EMERGENCY RESPONSE PLANS and PREPAREDNESS in SUPPORT of NUCLEAR POWER PLANTS

Criteria for Emergency Planning in an Early Site Permit Application

# DRAFT

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ABSTRACT

The Nuclear Regulatory Commission (NRC) and the Federal Emergency Management Agency (FEMA) have added Supplement 2 to NUREG-0654/FEMA-REP-1, Revision 1, to provide guidance for the development, review, and approval of radiological emergency information and plans submitted with an early site permit application under Subpart A of 10 CFR Part 52.

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# CRITERIA FOR EMERGENCY PLANNING IN AN EARLY SITE PERMIT APPLICATION

## I. INTRODUCTION

### A. Purpose and Use of Document

The Nuclear Regulatory Commission (NRC) and the Federal Emergency Management Agency (FEMA) have added Supplement 2 to NUREG-0654/FEMA-REP-1, Revision 1, to provide guidance for the development, review, and approval of radiological emergency preparedness information and plans submitted with an early site permit (ESP) application under Subpart A of 10 CFR Part 52. Because this document is a companion to NUREG-0654/FEMA-REP-1, Revision 1 and Supplement 1, only those changes relevant to an ESP application have been made. The extent of the use of this document, as well as the existing NUREG-0654/FEMA-REP-1, Revision 1, will depend on the extent of emergency planning information in the ESP application. In the absence of State and local participation, an applicant may use this document in conjunction with Supplement 1 of NUREG-0654/FEMA-REP-1, Revision 1, to prepare a utility offsite emergency plan as part of its ESP application. A notice has been provided in the Federal Register (FR) to announce the availability of this document.

### B. Authorities

This document facilitates the implementation of Subpart A of 10 CFR Part 52 (Title 10 of the Code of Federal Regulations), effective May 18, 1989 (54 FR 15372), for ESPs. Section 52.18 of that rule describes standards for review of ESP applications and provides for NRC consultation with FEMA in the review effort.

This document is consistent with the provisions of the FEMA/NRC Memorandum of Understanding dated April 18, 1985 (50 FR 15485), wherein the parties agreed that they would cooperate in radiological emergency planning matters and that FEMA would review available offsite plans and provide its findings and determinations.[1]

### C. Scope

This document provides guidance for ESP applicants and NRC and FEMA reviewers in the preparation and evaluation of emergency planning aspects of ESP applications. It suggests an approach for an applicant to identify physical characteristics that are unique to a proposed site that could pose significant impediments to the development of emergency plans. It also helps the applicant for an ESP determine the exact sizes of the emergency planning zones

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[1]Assessments of offsite plans may be based on State and local government plans submitted to FEMA under its rule (44 CFR Part 350) and, as noted in 44 CFR 350.3(f), may also be based on plans currently available to FEMA or furnished to FEMA through the NRC/FEMA Steering Committee.

(EPZs) for the proposed site. Section II of NUREG-0654/FEMA-REP-1, Revision 1, has been revised to provide selected planning standards and evaluation criteria applicable to the major features of the emergency plans. The revised standards and criteria reflect the particular conditions of the ESP phase of the licensing process. The document references other sources of guidance for preparing complete and integrated plans. Finally, this document emphasizes the need for an applicant under 10 CFR 52.17(b)(3) to make contacts and arrangements with local, State, and Federal agencies with emergency planning responsibilities and, in some circumstances, to obtain certifications from these agencies.

D. Definitions

The following definitions apply to the use of this document:

1. Early site permit. Commission approval issued pursuant to Subpart A of 10 CFR Part 52 for a site or sites for one or more nuclear power facilities.

2. Significant impediment. A physical characteristic or combination of physical characteristics that would pose major difficulties for an evacuation or the taking of other protective actions as addressed in Section II of this document.

3. Emergency plans. The radiological emergency response plans of the applicant, State, and local governments or, in the absence of participation by State and local governments, an applicant (utility) only plan.

4. Major features of the emergency plans. These include the exact sizes of the EPZs and other features as described in Section V of this document.

5. Combined license. A combined construction and operating license with conditions for a nuclear power facility issued pursuant to Subpart C of 10 CFR Part 52.

E. Emergency Planning Provisions of the Rule

The Commission promulgated 10 CFR Part 52 to provide for the issuance of ESPs, standard design certifications, and combined construction permits and operating licenses for nuclear power reactors. Subpart A of the rule sets out the requirements and procedures applicable to Commission issuance of ESPs for approval of a site or sites for one or more nuclear power facilities separate from the filing of an application for a construction permit or combined license for such a facility. Subpart A includes provisions for addressing emergency planning issues before any construction permit or combined license proceeding. These provisions follow.

An ESP application must identify physical characteristics unique to the proposed site, such as egress limitations from the area surrounding the site, that could pose a significant impediment to the development of emergency plans. The Commission shall determine, after consultation with FEMA, whether the information required of the applicant shows that there is no significant impediment to the development of emergency plans.

After meeting this mandatory requirement, the applicant may also exercise one of the two following options: (1) Propose major features of the emergency plans such as the exact sizes of the EPZs, for review and approval by NRC in consultation with FEMA in the absence of complete and integrated emergency plans or (2) propose complete and integrated plans for review and approval by the NRC in consultation with FEMA in accordance with the applicable provisions of 10 CFR 50.47. The Commission shall determine after consultation with FEMA if any major features of emergency plans submitted by the applicant under Option 1 are acceptable, and whether any emergency plans submitted by the applicant under Option 2 provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency.

For the mandatory requirement and Option 1, the application must include a description of contacts and arrangements made with local, State, and Federal agencies with emergency planning responsibilities. Under Option 2, the applicant shall make good faith efforts to obtain from the same government agencies certifications that (1) the proposed emergency plans are practicable; (2) these agencies are committed to participating in any further development of the plans, including any required field demonstrations; and (3) that these agencies are committed to executing their responsibilities under the plans in the event of an emergency. The application must contain any certifications that have been obtained. If these certifications cannot be obtained, the application must contain information, including a utility plan, sufficient to show that the proposed plans provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency at the site.

Subpart B of 10 CFR Part 52 addresses the requirements and procedures applicable to standard design certifications. Emergency planning requirements under Subpart B are limited primarily to the specification of an onsite Technical Support Center and an onsite Operational Support Center within the design bases of the standard plant design. Subpart C of the rule addresses the requirements and procedures applicable to the issuance of a combined license for a nuclear power facility. Under Subpart C, the application must contain emergency plans which meet the emergency planning standards of 10 CFR 50.47 and the requirements of Appendix E to Part 50 and thus provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency at the site. If the application for a combined license references an ESP, the application may incorporate by reference emergency plans, or major features of emergency plans, approved in conjunction with the issuance of the permit.

Guidance for the preparation and evaluation of emergency plans submitted in support of an ESP under Subpart A of 10 CFR 52 is provided in this document.

## II. EARLY SITE PERMITS - IDENTIFICATION OF PHYSICAL CHARACTERISTICS

The ESP application must identify physical characteristics unique to the proposed site, such as egress limitations from the area surrounding the site, that could pose a significant impediment to the development of emergency plans. An ESP applicant may identify such unique physical characteristics by performing a preliminary analysis of the time required to evacuate various sectors and distances within the plume exposure pathway EPZ for transient and permanent populations, noting major impediments to the evacuation or the taking of other protective actions.

### A. Evacuation Time Estimate Analysis

The evacuation time estimate (ETE) analysis is an emergency planning tool that can be used to assess, in an organized and systematic fashion, the feasibility of developing emergency plans for a site. Guidance on performing an ETE analysis is given in Appendix 4 to NUREG-0654/FEMA-REP-1, Revision 1. The ETE analysis should include a map showing the proposed site and plume exposure pathway EPZ. The map should identify transportation networks, topographical features, and political boundaries. The boundaries of the EPZ, as well as the evacuation sub-areas within the EPZ, should be based on factors such as current and projected demography, topography, land characteristics, access routes, and jurisdictional boundaries.

The ETE analysis should include an estimate of the number of people to be evacuated. Permanent residents, transients, and persons in special facilities should be considered in the population estimate. Special facility residents include those confined to institutions such as hospitals, nursing homes, and prisons. The school population also should be evaluated in the special facility segment. The ETE analysis should include a complete review and description of the road network in the proposed site area. The assumptions for determining the number of vehicles should be provided as well as the methodology for determining the transport-dependent population. Travel times and potential locations for serious congestion along the evacuation routes should be analyzed. Normal and adverse weather conditions, such as flooding, snow, ice, fog, or rain, should be considered in the ETE analysis. Additional guidance on performing an ETE analysis is given in NUREG/CR-4831, "State of the Art in Evacuation Time Estimate Studies for Nuclear Power Plants," March 1992.

Such an ETE analysis would serve to demonstrate if any physical characteristics or combination of physical characteristics of the site, in particular egress limitations, could pose impediments to the development of emergency plans. It is important to note that the value of the ETE analysis is in the methodology required to perform the analysis rather than in the calculated ETE times. While lower ETEs may reflect favorable site characteristics from an emergency planning standpoint, there is no minimum required evacuation time in the regulations which a licensee or an applicant has to meet. The Commission confirmed this in CLI-86-13, 24 NRC 22 (1986), when it stated that "our emergency planning requirements do not require that an adequate plan achieve a

preset....minimum evacuation time for the plume exposure pathway emergency planning zone in the event of a serious accident." Accordingly, the ETE analysis should not focus on the numerical time estimates but on the site factors that are considered to be impediments to emergency planning and preparedness. The reasons should be given for ETEs that appear unduly high. Any major difficulties for an evacuation or the taking of other protective actions such as sheltering in the plume EPZ should be discussed.

B. Contacts and Arrangements

The ESP application must include a description of contacts and arrangements made with local, State, and Federal agencies with emergency planning responsibilities. The descriptions should include the name and location of the organization contacted, the title and/or position of the person(s) contacted, and the role of the organization in emergency planning.

## III. EARLY SITE PERMITS - MAJOR FEATURES OF THE EMERGENCY PLANS

In addition to the mandatory requirement of identifying the physical characteristics unique to the proposed site that could pose impediments to the development of emergency plans, the ESP applicant may include proposed major features of the emergency plans, such as the exact sizes of the emergency planning zones (EPZs), for review and approval by the NRC in consultation with FEMA in the absence of complete and integrated plans.

### A. Emergency Planning Zones

An ESP applicant that chooses the option of proposing major features of the emergency plans (i.e., applicant, local and state plans), should give special emphasis to the exact sizes of the EPZs. Generally, the plume exposure pathway EPZ for nuclear power plants with an authorized power level greater than 250 MW thermal consists of an area about 10 miles in radius, and the ingestion pathway EPZ consists of an area about 50 miles in radius. The exact size and configuration of the EPZs surrounding a particular nuclear power reactor should be determined in relation to local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries [10 CFR 50.47 (c)(2)]. Plume exposure pathway EPZ boundaries that run through the middle of schools or hospitals, or that arbitrarily carve out small portions of governmental jurisdictions should be avoided [CLI 89-12, 26 NRC 383 (1987)]. Additional information concerning EPZs is contained in Section I.D.2 of NUREG-0654/FEMA-REP-1, Revision 1, and 44 CFR 350.7(b).

### B. Planning Standards and Evaluation Criteria

An ESP application that includes the major features of emergency plans will be evaluated against the selected and modified emergency planning standards and evaluation criteria from Section II of NUREG-0654/FEMA-REP-1, Revision 1. These standards and criteria are specified in Section V of this document and have been selected to

- highlight the need for cooperation among the applicant, local, State, and Federal agencies required by 10 CFR 52.17(b)(3)
- address potential emergency planning issues early in the licensing process before large commitments of resources are made
- reflect sensitivity to the fact that an ESP applicant will not have information and resources, or should not be expected to expend large resources, on aspects of emergency planning and preparedness that will be required, and best addressed, at the combined license stage
- reflect sensitivity to the relatively long time (10 to 20 years) that could elapse between the granting of an ESP and the application for a combined license

In addition, the standards and criteria referring to facilities, systems, and equipment have been modified to require only descriptions rather than in-place capability.

The modifications to the emergency planning standards and evaluation criteria in Section V of this document apply only to an early site permit application. All of the planning standards of 10 CFR 50.47(b), as supported in NUREG-0654/ FEMA-REP-1, Revision 1, will have to be met before the issuance of an operating license under 10 CFR 50.57 or a combined license under Subpart C of 10 CFR 52. In addition, for the first reactor at a site, Appendix E of 10 CFR 50 requires that a full-participation exercise be conducted within 2 years before the NRC issues an operating license for full power (authorizing operation above 5 percent of rated power). Since this exercise would be included in the inspections, tests, and analyses required for a combined license, it would have to be satisfied before fuel loading for a combined license.

C. Contacts and Arrangements

The ESP application under this option of the rule must include a description of contacts and arrangements made with local, State, and Federal agencies with emergency planning responsibilities. The descriptions should include the name and location of the organization contacted, the title and/or position of the person(s) contacted, and the role of the organization in emergency planning. Additional guidance concerning contacts and arrangements for this option of the rule are specified in the evaluation criteria in Section V of this document.

## IV. EARLY SITE PERMITS - COMPLETE AND INTEGRATED PLANS

### A. Planning Standards and Evaluation Criteria

An applicant for an ESP may propose complete and integrated emergency plans for review and approval by the NRC, in consultation with FEMA, in accordance with the applicable provisions of 10 CFR 50.47. The planning standards and evaluation criteria for the preparation and evaluation of these plans are as shown in NUREG-0654/FEMA-REP-1, Revision 1.

### B. Certifications from Governmental Agencies

The ESP applicant proposing complete and integrated emergency plans must make good faith efforts to obtain certifications from the local, State, and Federal agencies with emergency planning responsibilities. The certifications should state

- the agencies' acceptance of the proposed emergency plans as practicable
- the agencies' commitment to participate in any further development of the plan, including any required field demonstrations
- the agencies' commitment to execute their responsibilities under the plans in the event of an emergency

### C. Non-Participation of Government Agencies

If the ESP applicant is unable to obtain certifications from local, State and Federal agencies with emergency planning responsibilities, it should prepare an offsite plan which will be evaluated using NUREG-0654/FEMA-REP-1, Revision 1, Supplement 1.

## V. PLANNING STANDARDS AND EVALUATION CRITERIA FOR MAJOR FEATURES OF THE EMERGENCY PLAN

An ESP applicant may use the planning standards and evaluation criteria given in this section if it chooses the option of proposing major features of the emergency plans. The standards and criteria have been developed to reflect the particular conditions of an ESP based on the planning standards and evaluation criteria of NUREG-0654/FEMA-REP-1, Revision 1.

Comparisons to the planning standards and evaluation criteria of NUREG-0654/FEMA-REP-1, Revision 1, are given below.

1. The following planning standards and related evaluation criteria of NUREG-0654 were deemed inappropriate for the ESP application phase:

   M. Recovery and Reentry Planning and Postaccident Operations
   N. Exercises and Drills

2. The following planning standards are comparable to those in NUREG-0654 but are more relevant to the ESP application phase, with certain evaluation criteria (EC) not included as shown in the second column.

| STANDARD | EC NOT INCLUDED |
|---|---|
| A. Assignment of Responsibility (Organization Control) | 1b, c, d, e, and 4 |
| B. Onsite Emergency Organization | 1, 2, 3, 4, 5, 7, and 8 |
| C. Emergency Response Support and Resources | 1a, b, c, 2a, and b |
| D. Emergency Classification System | 2 and 4 |
| E. Notification Methods and Procedures | 3, 4, 5, and 7 |
| F. Emergency Communications | 1a, 1d, 1f, and 3 |
| G. Public Education and Information | 2, 3, and 4 |
| H. Emergency Facilities and Equipment | 4, 5, 6, 7, 8, 9, 10, 11, and 12 |
| I. Accident Assessment | 1, 2, 3, 4, 6, 8, 9, and 10 |
| J. Protective Response | 1, 3, 4, 5, 6, 9, 10f, and 11 |
| K. Radiological Exposure Control | 3b, 6, and 7 |

| STANDARD, cont. | EC NOT INCLUDED, cont. |
|---|---|
| L. Medical and Public Health Support | 2 and 4 |
| O. Radiological Emergency Response Training | 1, 2, 3, 4e, 4i, and 5 |
| P. Responsibility for the Planning Effort | 6, 7, 9, and 10 |

3. The retained evaluation criteria have been modified to make them consistent with changes in related planning standards and more appropriate to the ESP application phase.

4. A new evaluation criterion has been added: EC 4, under Planning Standard C, Emergency Response Support and Resources.

5. Under Applicability and Cross Reference to Plans, the term "ESP Applicant" has been substituted for the term "Licensee."

The planning standards and evaluation criteria as modified are as follows.

A. Assignment of Responsibility (Organization Control)

Planning Standard

Primary responsibilities are identified for emergency response by the Early Site Permit (ESP) applicant and by State and local organizations within the Emergency Planning Zones (EPZs).

| Evaluation Criterion | Applicability and Cross Reference to Plans | | |
|---|---|---|---|
| | ESP Applicant | State | Local |
| 1. Each application shall identify the State, local, Federal and private sector organizations (including utilities) that are intended to be part of the overall response organization for EPZs. | X | X | X |
| 2. a. Each organization shall identify the functions and responsibilities for major elements of emergency response, such as Command and Control, Alerting and Notification, Communications, Public Information, Accident Assessment, Public Health and Sanitation, Social Services, Fire and Rescue, Traffic Control, Emergency Medical Services, Law Enforcement, Transportation, Protective Response, and Radiological Exposure Control. The description of these functions may be shown in a table of primary and support responsibilities. | | X | X |
| b. Each application shall contain (by reference to specific acts, codes or statutes) the legal basis for such authorities. | | X | X |
| 3. Each application shall include a description of contacts and arrangements pertaining to the concept of operations developed between Federal, State, and local agencies and other support organizations having an emergency response role within the EPZs. Any written letters of agreement should be included. A signature page in the application may be appropriate for some organizations to signify their agreement. | X | X | X |

B. Onsite Emergency Organizations

Planning Standard

Interfaces among various onsite response activities and offsite support and response activities are identified.

| Evaluation Criterion | Applicability and Cross Reference to Plans | | |
|---|---|---|---|
| | ESP Applicant | State | Local |
| 1. Each applicant shall identify the interfaces between and among the onsite functional areas of emergency activity, local services support, and State and local government response organization. This may be illustrated in a block diagram. | X | | |
| 2. Each applicant shall identify the services to be provided by local agencies for handling emergencies (e.g., police, ambulance, medical, hospital, and fire-fighting organizations). A description of the arrangements involving these services shall be included in the plan. Any written letters of agreement should also be included. | X | | |

C. Emergency Response Support and Resources

Planning Standard

Arrangements for requesting assistance resources are described, and organizations capable of augmenting the planned response are identified.

| Evaluation Criterion | Applicability and Cross Reference to Plans | | |
|---|---|---|---|
| | ESP Applicant | State | Local |
| 1. The Federal Government maintains in-depth capability to assist licensees, States and local governments through the Federal Radiological Emergency Response Plan. Each State and ESP applicant shall make provisions for requesting Federal assistance. | X | X | |
| 2. Each organization shall identify radiological laboratories and their general capabilities and expected availability to provide radiological monitoring and analyses services during an emergency. | X | X | X |
| 3. Each organization shall identify nuclear and other facilities and organizations that can be relied on to provide assistance in an emergency. | X | X | X |
| 4. Each application shall include a description of the contacts and arrangements made with the response organizations identified above. | X | X | X |

D. Emergency Classification System

Planning Standard

A standard emergency classification scheme is specified.

| Evaluation Criterion | Applicability and Cross Reference to Plans | | |
|---|---|---|---|
| | ESP Applicant | State | Local |
| 1. An emergency classification scheme as set forth in Appendix 1 of NUREG-0654/FEMA-REP-1, Revision 1, or Regulatory Guide 1.101, Revision 3, must be established by the ESP applicant but need not include plant-specific initiating conditions. | X | | |
| 2. Each State and local organization shall establish an emergency classification level scheme consistent with that established by the ESP applicant. | | X | X |

E. Notification Methods and Procedures

Planning Standard

The means are described for notification by the ESP applicant of State and local response organizations, and for notification of emergency personnel and the populace within the plume exposure pathway EPZ.

| Evaluation Criterion | Applicability and Cross Reference to Plans | | |
|---|---|---|---|
| | ESP Applicant | State | Local |
| 1. Each organization shall describe mutually agreeable bases for notification of response organizations consistent with the emergency classification scheme set forth in Appendix 1 of NUREG-0654/FEMA-REP-1, Revision 1. | X | X | X |
| 2. Each organization shall describe a method for alerting, notifying, and mobilizing emergency response personnel. | X | X | X |
| 3. Each organization shall describe the administrative and physical means for notifying and promptly instructing the public within the plume exposure pathway EPZ. | X | X | X |

F. Emergency Communications

Planning Standard

Provisions are described for prompt communications among principal response organizations to emergency personnel and to the public.

| Evaluation Criterion | Applicability and Cross Reference to Plans | | |
|---|---|---|---|
| | ESP Applicant | State | Local |
| 1. The communication plans for emergencies shall describe: | | | |
| a. provision for communications with contiguous State/local governments within the EPZ | X | X | X |
| b. provision for communications as needed with Federal emergency response organizations | X | X | X |
| c. provision for alerting and activating emergency personnel in each response organization | X | X | X |
| 2. The communication arrangement for fixed and mobile medical support facilities shall be described. | X | X | X |

G. Public Education and Information

Planning Standard

An emergency information program for the public and news media is described to address: (1) the provision of information to the public, on a periodic basis, on how they will be notified and what initial actions should be taken in an emergency and (2) the means for acquainting the news media with emergency information.

| Evaluation Criterion | Applicability and Cross Reference to Plans | | |
|---|---|---|---|
| | ESP Applicant | State | Local |
| 1. Each organization shall describe a program to provide a coordinated dissemination of information to the public on a periodic basis (at least annually) regarding how they will be notified and what their actions should be in an emergency. This program should include information on:<br>• educational information on radiation<br>• contact for additional information<br>• protective measures, e.g., evacuation routes and relocation centers, sheltering, respiratory protection, radioprotective drugs<br>• special needs of the handicapped and the transient population<br>Means for accomplishing this dissemination may include, but are not necessarily limited to: information in the telephone book, periodic information in utility bills, posting in public areas, and publications distributed on an annual basis. | X | X | X |
| 2. Each organization shall describe a program for acquainting news media on a periodic basis (at least annually) with the emergency plans, information concerning radiation, and points of contact for release of public information in an emergency. | X | X | X |

H. Emergency Facilities and Equipment

Planning Standard

Adequate emergency facilities and equipment to support the emergency response are described.

| Evaluation Criterion | Applicability and Cross Reference to Plans | | |
|---|---|---|---|
| | ESP Applicant | State | Local |
| 1. Each applicant shall describe a technical support center and an onsite operations support center in accordance with NUREG-0696.[2] | X | | |
| 2. Each applicant shall describe an emergency operations facility from which evaluation and coordination of all licensee activities related to an emergency is to be carried out and from which the licensee shall provide information to Federal, State and local authorities responding to radiological emergencies in accordance with NUREG-0696. | X | | |
| 3. Each offsite organization shall describe an emergency operations center for use in directing and controlling response functions. | | X | X |

[2] NUREG-0696, "Functional Criteria for Emergency Response Facilities," February 1981.

I. Accident Assessment

Planning Standard

Adequate methods, systems and equipment are described for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition.

| Evaluation Criterion | Applicability and Cross Reference to Plans | | |
|---|---|---|---|
| | ESP Applicant | State | Local |
| 1. Each applicant shall provide a description of the contacts and arrangements made with offsite organizations for acquiring and evaluating meteorological information. The applicant shall describe how suitable meteorological data will be made available to the State. | X | | |
| 2. Each organization shall describe the contacts and arrangements made for field monitoring within the plume exposure EPZ. | X | X | X |
| 3. Contacts and arrangements to locate and track the airborne radioactive plume, using either or both Federal and State resources, shall be described. | X | X | |

J. Protective Response

Planning Standard

A range of protective actions is described for the plume exposure pathway EPZ for the public and emergency workers. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, and protective actions for the ingestion exposure pathway EPZ appropriate to the locale are described.

| Evaluation Criterion | Applicability and Cross Reference to Plans | | |
|---|---|---|---|
| | ESP Applicant | State | Local |
| 1. Each applicant shall describe the evacuation routes and transportation for onsite individuals to some suitable off-site location, including alternatives for inclement weather, high traffic density, and specific radiological conditions. | X | X | X |
| 2. Each applicant shall describe a mechanism for recommending protective actions to the appropriate State and local authorities in accordance with the Manual of Protective Action Guides and Protective Actions for Nuclear Incidents (EPA 400-R-92-001). | X | | |
| 3. Each applicant must prepare time estimates for evacuation within the plume exposure EPZ (see Section II of this document). These shall be in accordance with Appendix 4 of NUREG-0654/FEMA-REP-1, Revision 1. | X | | |

J. Protective Response (continued)

| Evaluation Criterion | Applicability and Cross Reference to Plans | | |
|---|---|---|---|
| | ESP Applicant | State | Local |
| 4. Each organization's concept for implementing protective measures for the plume exposure pathway should be described including items such as those listed below, if available. Where appropriate, a description of contacts and arrangements made with offsite agencies with emergency planning responsibilities should be included. | | | |
| a. maps showing evacuation routes, evacuation areas, shelter areas, and relocation centers in host areas | X | X | X |
| b. maps showing population distribution around the site This shall be by evacuation areas. (Each applicant shall also present the information in a sector format) | X | X | X |
| c. proposed means for notifying all segments of the transient and resident population | X | X | X |
| d. proposed means for protecting those persons whose mobility may be impaired (e.g. institution or other confinement) | | X | X |
| e. proposed means for the use of radioprotective drugs for emergency workers and institutionalized persons within the plume exposure EPZ whose immediate evacuation may be infeasible or very difficult | | | |
| f. proposed means of relocation | | X | X |

J. Protective Response (continued)

| Evaluation Criterion | Applicability and Cross Reference to Plans | | |
|---|---|---|---|
| | ESP Applicant | State | Local |
| g. potential relocation centers in host areas which are at least 5 miles, and preferably 10 miles, beyond the boundaries of the plume exposure emergency planning zone | | X | X |
| h. projected traffic capacities of evacuation routes under emergency conditions | | X | X |
| i. control of access to evacuated areas and organization responsibilities for such control | | X | X |
| j. identification of and means for dealing with potential impediments (e.g., seasonal impassability of roads) to use of evacuation routes and contingency measures | | X | X |
| k. time estimates for evacuation of various sectors and distances based on a dynamic analysis (time-motion study under various conditions) for the plume exposure pathway EPZ (See Appendix 4 of NUREG-0654/FEMA-REP-1, Revision 1, and Section II of this document) | | X | X |
| l. the bases for the choice of recommended protective actions from the plume exposure pathway during emergency conditions. This shall include expected local protection afforded in residential units or other shelter for direct and inhalation exposure, as well as evacuation time estimates | | X | X |
| 5. Each organization shall describe the means for registering and monitoring evacuees at reception centers in host areas. | | X | X |

K. Radiological Exposure Control

Planning Standard

Means are described for controlling radiological exposures to emergency workers in an emergency.

| Evaluation Criterion | | Applicability and Cross Reference to Plans | | |
|---|---|---|---|---|
| | | ESP Applicant | State | Local |
| 1. | Each applicant shall describe guidelines on dose limits for | | | |
| | a. removal of injured persons | X | | |
| | b. undertaking corrective actions | X | | |
| | c. performing assessment actions | X | | |
| | d. performing field radiological measurements in the plume EPZ | X | | |
| | e. providing first aid | X | | |
| | f. performing personnel decontamination | X | | |
| | g. providing ambulance service | X | | |
| | h. providing medical treatment services | X | | |
| 2. | Each applicant shall describe an onsite radiation protection program to be implemented during emergencies, including methods to implement dose limits. General guidance on dose limits for workers performing emergency services can be found in EPA 400-R-92-001.[3] | X | | |
| 3. | a. Each organization shall describe how they would determine the doses received by emergency personnel involved in any nuclear accident, including volunteers. | X | X | X |

[3] EPA 400-R-92-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," May 1992.

K. Radiological Exposure Control (continued)

| Evaluation Criterion | Applicability and Cross Reference to Plans | | |
|---|---|---|---|
| | ESP Applicant | State | Local |
| b. Each organization shall describe how they would acquire and distribute dosimeters, both direct-reading and permanent record devices. | X | X | X |
| 4. Each organization shall describe a decision chain for authorizing emergency workers to incur exposures in excess of the EPA dose limits for workers performing emergency services. | X | X | X |
| 5. a. Each organization, as appropriate, shall specify action levels for determining the need for decontamination of emergency workers, equipment and vehicles, and the general public and their possessions. | X | X | X |
| b. Each organization, as appropriate, shall describe a means for radiological decontamination of emergency personnel wounds, supplies, instruments, and equipment. | X | X | X |

L. Medical and Public Health Support

Planning Standard

Contacts and arrangements are described for medical services for contaminated injured individuals.

| Evaluation Criterion | [3]Applicability and Cross Reference to Plans | | |
|---|---|---|---|
| | ESP Applicant | State | Local |
| 1. Each organization shall describe the contacts and arrangements made for local and backup hospital and medical services having the capability for evaluation of radiation exposure and uptake. | X | X | X |
| 2. Each State shall develop lists indicating the location of public, private, and military hospitals and other emergency medical services facilities within the State or contiguous States considered capable of providing medical support for any contaminated injured individual. The listing shall include the name, location, type of facility and capacity, and any special radiological capabilities. Contacts and arrangements made in developing this list should be described. | | X | |

M. Recovery and Reentry Planning and Postaccident Operations (Not applicable to ESPs)

N. Exercises and Drills (Not applicable to ESPs)

O. Radiological Emergency Response Training

Planning Standard

A radiological emergency response training program is described for those who may be called on to assist in an emergency.

| Evaluation Criterion | Applicability and Cross Reference to Plans | | |
|---|---|---|---|
| | ESP Applicant | State | Local |
| 1. Each organization shall describe a training program for instructing and qualifying personnel who will implement radiological emergency response plans. Specialized initial training and periodic retraining shall be provided in the following categories: | | | |
| a. directors or coordinators of the response organizations | X | X | X |
| b. personnel responsible for accident assessment | X | X | X |
| c. radiological monitoring teams and radiological analysis personnel | X | X | X |
| d. police, security and fire fighting personnel | X | X | X |
| f. first aid and rescue personnel | X | X | X |
| g. local support services personnel including civil defense/emergency service personnel | X | X | X |
| h. medical support personnel | X | X | X |
| i. personnel responsible for transmission of emergency information and instructions | X | X | X |

P. Responsibility for the Planning Effort: Development, Periodic Review and Distribution of Emergency Plans

Planning Standard

Responsibilities are established for plan development and review and for distribution of emergency plans, and training is described for planners.

| Evaluation Criterion | | Applicability and Cross Reference to Plans | | |
|---|---|---|---|---|
| | | ESP Applicant | State | Local |
| 1. | Each organization shall provide for the training of individuals responsible for the planning effort. | X | X | X |
| 2. | Each organization shall identify by title the individual with the overall authority and responsibility for radiological emergency response planning. | X | X | X |
| 3. | Each organization shall designate an Emergency Planning Coordinator with responsibility for the development and updating of emergency plans and coordination of these plans with other response organizations. | X | X | X |
| 4. | Each organization shall update its plan and agreements as needed. | X | X | X |
| 5. | The emergency response plans and approved changes to the plans shall be forwarded to all organizations and appropriate individuals with responsibility for implementation of the plans. Revised pages shall be dated and marked to show where changes have been made. | X | X | X |
| 6. | Each plan shall contain a specific table of contents. Plans submitted for review should be cross-referenced to the criteria in this supplement. | X | X | X |

# CERTIFIED STANDARD DESIGN RELATED ITAAC

ENCLOSURE 2

## (Example)

| CERTIFIED DESIGN COMMITMENT | INSPECTIONS, TESTS, ANALYSES | ACCEPTANCE CRITERIA |
|---|---|---|
| 1. Provide an on-site Technical Support Center (TSC). | 1. Inspect the physical location of the TSC. | 1. a. The TSC is located within the Protected Area.<br><br>b. The TSC is located in close proximity to the control room. The walking time from the TSC to the control room does not exceed 2 minutes.<br><br>c. The size of the TSC is sufficient to accommodate a minimum of 20 licensee and 5 NRC persons. (Provide minimum of 2.32 $m^2$ per person and minimum overall space of 174.18 $m^2$.)<br><br>d. The TSC is structurally built in accordance with the Uniform Building Code.<br><br>e. The TSC shall provide radiation protection to assure that any person working in the TSC will not exceed 5 rem whole body or equivalent to any part of the body for the duration of an accident. (Reference ITAAC 3, Table 3.7a: Plant Shielding Design ABWR Design Document)<br><br>f. Provisions have been made in the TSC for:<br><br>(1) Reliable Communications to the OSC, Control Room, and the EOF.<br><br>(2) Data retrieval and storage.<br><br>(3) Environmentally controlled (HVAC). - The TSC ventilation |

| CERTIFIED DESIGN COMMITMENT | INSPECTIONS, TESTS, ANALYSES | ACCEPTANCE CRITERIA |
|---|---|---|
| | | system shall function in a manner comparable to the control room ventilation system. It includes high-efficiency particulate air (HEPA) and charcoal filters as a minimum. It need not be seismic Category I qualified, redundant, instrumented in the control room, or automatically activated).<br><br>(4) Storage facilities for plant specific information such as basis document, drawings, and procedures without leaving the TSC/CR ventilation space. |
| 2. Provide an on-site Operational Support Center (OSC). | 2. Inspect the physical location of the OSC. | 2. a. The OSC is located onsite.<br><br>b. Provisions are made to obtain reliable communications to the control room, the TSC, and the EOF. |
| 3. Provide on-site Support Facilities. | 3. Inspect the physical location and operation of on-site support facilities. | 3. a. The onsite Personnel Decontamination Facility (PDF) and the onsite Laboratory are located to facilitate support following an accident.<br><br>b. The design basis for plant equipment for reactor coolant and containment atmosphere sampling and analysis must assume that it is possible to obtain and analyze a sample without radiological exposures to any individual exceeding the criteria of GDC19 (Appendix A, 10 CFR Part 50) (i.e., 5 rem whole body 75 rem extremities). |

# COMBINED LICENSE ITAAC RELATED TO EMERGENCY PLANNING
## (EXAMPLE)

| COMBINED LICENSE COMMITMENT | INSPECTIONS, TESTS, ANALYSES | ACCEPTANCE CRITERIA |
|---|---|---|
| 1. Assign emergency planning responsibilities | 1. Verify that primary responsibilities for emergency response by the nuclear facility licensee and by State and local organizations within the Emergency Planning Zones have been assigned in each plan. The emergency responsibilities of the various supporting organizations have been specifically established. Each principal response organization is provided with adequate staffing to respond. | 1. a. The plan identifies the State, local, Federal and private sector organizations (including utilities), that are intended to be part of the overall response organization for Emergency Planning Zones.<br><br>b. The plan for each organization and suborganization having an operational role clearly specifies its scope of operations and its relationship to the total effort.<br><br>c. The plan for each organization identifies a specific individual by title who shall be in charge of the emergency response.<br><br>d. Adequate staffing is provided for each principle organization for 24-hours per day emergency response for a protracted (30 day) period, including 24-hour per day manning of communication links. The individual in each principle organization who will be responsible for assessing continuity of resources (including technical, administrative and material) is specified by title.<br><br>e. The plan for each offsite organization specifies the |

| COMBINED LICENSE COMMITMENT | INSPECTIONS, TESTS, ANALYSES | ACCEPTANCE CRITERIA |
|---|---|---|
| | | functions and responsibilities for major elements and key individuals by title, of emergency response, including the following: Command and Control, Alerting and Notification, Communications, Public Information, Accident Assessment, Public Health and Sanitation, Social Services, Fire and Rescue, Traffic Control, Emergency Medical Services, Law Enforcement, Transportation, Protective Response (including authority to request Federal assistance and to initiate other protective actions), and Radiological Exposure Control. The description of these functions includes a clear and concise summary such as a table of primary and support responsibilities using the agency as one axis and the function as another. |
| | | f. Each plan contains (by reference to specific acts, codes, or statutes) the legal basis for authorities in letter e above.<br><br>Each plan includes written agreements referring to the concept of operations developed between Federal, State, and local agencies and other support organizations having an |

| COMBINED LICENSE COMMITMENT | INSPECTIONS, TESTS, ANALYSES | ACCEPTANCE CRITERIA |
|---|---|---|
| | | emergency response role within the Emergency Planning Zones. The agreements identify the emergency measures to be provided and the mutually acceptable criteria for their implementation, and specify the arrangements for exchange of information. These agreements may be provided in an appendix to the plan or the plan itself may contain descriptions of these matters and a signature page in the plan may serve to verify the agreements. The signature page format is appropriate for organizations where response functions are covered by laws, regulations or executive orders where separate written agreements are not necessary. |

# COMBINED LICENSE ITAAC RELATED TO EMERGENCY PLANNING
## (EXAMPLE - CONTINUED)

| COMBINED LICENSE COMMITMENT | INSPECTIONS, TESTS, ANALYSES | ACCEPTANCE CRITERIA |
|---|---|---|
| 14. Conduct a full-participation exercise that tests as much of the licensee, State and local emergency plans as is reasonably achievable without mandatory public participation. Include participation by each State and local government within the plume exposure EPZ and each State within the ingestion exposure pathway EPZ. | a. Test the following aspects of licensee, State, and local emergency plans:<br>1) Adequacy and timeliness of implementing procedures and methods<br>2) Emergency equipment<br>3) Communication network | a. 1) The onsite and offsite portions of the exercise have been completed.<br>2) The specified aspects of emergency plans have been tested.<br>3) Deficiencies identified in the exercise which prevent a finding of reasonable assurance of adequate protective measures have been corrected.* |

* Deficiencies referred to in these acceptance criteria are defined in a June 17, 1993, Memorandum of Understanding between the NRC and the Federal Emergency Management Agency (58 FR 47996, September 14, 1993) as "...an observed or identified inadequacy of organizational performance in an exercise that could cause a finding that offsite emergency preparedness is not adequate to provide reasonable assurance that appropriate protective measures can be taken in the event of a radiological emergency to protect the health and safety of the public living in the vicinity of a nuclear power plant."

# ANNEX E

## ACRS REPORT ON FSER FOR EPRI REQUIREMENTS DOCUMENT FOR PASSIVE PLANT DESIGNS DECEMBER 23, 1993

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

December 23, 1993

The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Selin:

SUBJECT: ELECTRIC POWER RESEARCH INSTITUTE ADVANCED LIGHT WATER REACTOR UTILITY REQUIREMENTS DOCUMENT -- VOLUME III PASSIVE PLANTS

During the 402nd meeting of the Advisory Committee on Reactor Safeguards, October 7-8, 1993, we reviewed the staff Final Safety Evaluation Report (FSER) for Volume III of the Electric Power Research Institute (EPRI) Advanced Light Water Reactor (ALWR) Utility Requirements Document (URD) for Passive Plants. Our Subcommittee on Improved Light Water Reactors held a meeting on October 6, 1993, to review this subject. Our final deliberations on this matter occurred during our 404th meeting, December 9-11, 1993. 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.

In the early 1980s, EPRI established the ALWR program to support the United States utility industry efforts to ensure a viable nuclear power generation option for the 1990s and beyond. The overall objective was to establish utility industry policy along with top-tier technical and operational criteria for evolutionary and passive plant designs that would facilitate standardization and combined licensing. The intent of the program was to resolve as many of the policy, technical, and licensing issues as could be identified before specific plant designs were to be submitted, or approved. The remaining specific detailed technical and operational issues were to be resolved during consideration of detailed design information on specific plant design submittals. The program was to ensure that future nuclear power plants would be safer, simpler, more robust with greater margins, more easily operated and maintained, and more certain of being constructed and licensed without delays. The approach was to use utility experience to establish design philosophy, produce design criteria and guidance to achieve the objective, and to address the policies and regulations of the NRC.

The EPRI ALWR URD is a compendium of technical requirements for the design, construction, and performance of ALWR nuclear power plants for the 1990s and beyond. The URD consists of three volumes:

- Volume I, "ALWR Policy and Summary of Top-Tier Requirements," is a management-level synopsis of the URD, 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, "ALWR Evolutionary Plant," consists of 13 chapters and contains utility design requirements for evolutionary nuclear power plants.

- Volume III, "ALWR Passive Plant," consists of 13 chapters and contains utility design requirements for passive nuclear power plants.

We have followed the development of the EPRI ALWR program from its inception and offered suggestions regarding safety improvements on several occasions. We discussed development of the EPRI URD program and the NRC staff reviews during numerous Subcommittee and full Committee meetings. We previously presented our comments to the Commission pertaining to the FSER for Volume II by our report of August 18, 1992.

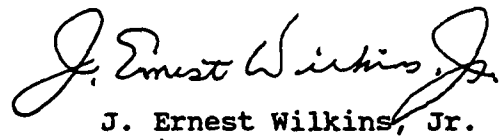
Volume III is similar to Volume II and many chapters are identical except for the features, requirements, and those policy, technical, and licensing issues unique to the passive plants. Although the Standard Review Plan (SRP) was used by the staff as guidance, the level of detail in the URD did not permit a verification of adequacy. (The SRP was written to support the review of the final safety analysis reports on specific plant designs for which a significant amount of design and construction information is normally available.) The staff conducted its review with the understanding that the EPRI design criteria would meet all current regulations, except where deviations were identified. The staff review of the URD focused on determining whether the EPRI criteria conflict with current regulatory requirements.

In addition, the staff identified a number of policy, technical, and licensing issues which needed resolution in order to complete its review of the ALWRs, including the URD. We provided comments on these issues by our referenced letters. The Commission considered the staff positions on twenty-one of the issues identified in SECY-93-087 pertaining to passive plants.

We believe that the staff has conducted a thorough and comprehensive review. We are in general agreement with the FSER pertaining

to Volume III and its conclusion that meeting the URD requirements could result in a reactor design that would not conflict with regulatory guidelines, and that would be responsive to various policy statements. Nevertheless, we are disappointed in the limited technical basis provided for several of the requirements relating to severe accidents – in particular hydrogen control, melt spreading and coolability, and fuel coolant interaction (steam explosion). In addition, we believe additional consideration should have been given to general design criteria for containment to withstand severe accident loads.

Sincerely,



J. Ernest Wilkins, Jr.
Chairman

References:

1. SECY-93-087, dated April 2, 1993, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs
2. SECY-92-172, dated May 12, 1992, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Final Safety Evaluation Report for Volume II of the Electric Power Research Institute's Advanced Light Water Reactor Requirements Document, including the following enclosures:
   - Draft Safety Evaluation Report for Volume I, "Program Summary of the NRC Review of the Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document," prepared by the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, dated May 1992
   - Final Safety Evaluation Report for Volume II, "NRC Review of the Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document for Evolutionary Plant Designs," prepared by the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, dated May 1992
3. Electric Power Research Institute, Advanced Light Water Reactor Utility Requirements Document, Volume II, "ALWR Evolutionary Plant," Chapters 1-13 through Revision 4, dated April 1992

4. Draft Commission Paper, undated, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs
5. Staff Requirements Memorandum dated July 21, 1993, from Samuel J. Chilk, Secretary, to James M. Taylor, Executive Director for Operations, Subject: SECY-93-087 - Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs
6. Letter dated November 10, 1993, from J. Ernest Wilkins, Jr., ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Draft Commission Paper, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-safety Systems in Passive Plant Designs"
7. Letter dated April 26, 1993, from Paul Shewmon, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs
8. Letter dated August 18, 1992, from David A. Ward, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Electric Power Research Institute Advanced Light Water Reactor Utility Requirements Document -- Volume II, Evolutionary Plants
9. Letter dated August 17, 1992, from David A. Ward, ACRS Chairman, to James M. Taylor, EDO, Subject: Issues Pertaining to Evolutionary and Passive Light-Water Reactors and Their Relationship to Current Regulatory Requirements
10. Letter dated May 13, 1992, from David A. Ward, ACRS Chairman, to James M. Taylor, EDO, Subject: Issues Pertaining to Evolutionary and Passive Light-Water Reactors and Their Relationship to Current Regulatory Requirements
11. Letter dated April 26, 1990, from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, NRC Chairman, Subject: Evolutionary Light-Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements

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U.S. NUCLEAR REGULATORY COMMISSION

# BIBLIOGRAPHIC DATA SHEET

*(See instructions on the reverse)*

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Project Number 669

11. ABSTRACT *(200 words or less)*
The staff of the U.S. Nuclear Regulatory Commission has prepared Volume 3 (Parts 1 and 2) of a safety evaluation report (SER), "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements - Passive 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 III of the Requirements Document for passive plant designs, which consists of 13 chapters and contains utility design requirements for nuclear power plants for which passive features will be used in their design (approximately 600 megawatts-electric).

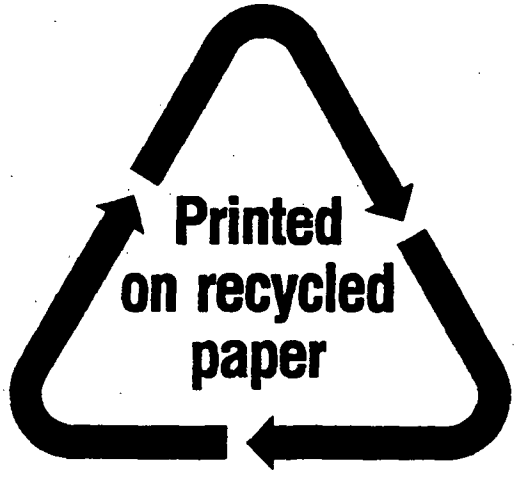
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