Draft Safety Evaluation Report

related to the final design approval and design certification of the Advanced Boiling Water Reactor

Docket No. 50-605

General Electric Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

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1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

This report is a safety evaluation report (SER) on the application by the General Electric Co, Inc. (GE) for final design approval (FDA) and design certification of its advanced boiling water reactor (ABWR) design.

This report was prepared by the U.S. Nuclear Regulatory Commission staff (NRC staff or staff) and summarizes the results of the staff's safety review of Chapters 4, 5, 6, and 17 of the ABWR Standard Safety Analysis Report (SSAR). The NRC Licensing Project Manager for the ABWR is Mr. Dino Scaletti. Mr. Scaletti may be reached by calling (301) 492-1104 or by writing to the: Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

This safety evaluation addresses those portion of the SSAR that were reviewed to Standard Review Plan (NUREG-0800) Sections 4, 5, 6, and 17. The application, which included Modules 4, 5, 6 and 15, was received in September 1987 and is being processed consistent with the commitments described in the Advanced Boiling Water Reactor Licensing Review Bases document dated August 1987 (see Appendix A to this SER). The licensing review bases (LRB) contain no new regulatory requirements. In certain key areas where Commission policies and staff positions are still under development, both the staff and GE have committed to implement acceptance criteria which, if satisfied by the ABWR standard design, would result in a licensable design. However, should substantial new information become available that results in the promulgation of new requirements by the NRC, they would be addressed during the course of the ABWR review. The regulations governing the submittal and review of standard designs under the "reference system" option are contained in 10 CFR 2.110 and were recently promulgated in 10 CFR Part 52, Subpart B, and Appendix O to 10 CFR Part 52. GE is responsible for the design of those systems that are within the ABWR scope and for sufficiently descriptive interface information so that individual applicants referencing the ABWR design need only describe the out-of-scope system, provide site-related information, and that conformance with the ABWR interface requirements. During the course of the review, the staff has audited and will perform other audits of detailed design analyses and interface

documentation that are retained in GE's offices in San Jose, California. These audits also are being performed on design information that has been developed for an application by Tokyo Electric Company to the Japanese Government (Ministry of International Trade and Industry). The Japanese application is for the construction and operation of two units of the ABWR design at the Kashiwazaki-Kariwa site on the Japan Sea. Differences between the ABWR design for U.S. certification and the Japanese design will be documented by GE in the ABWR SSAR. This information will be summarized in the ABWR SSAR and is discussed in the text of this report.

The staff is currently reviewing all the elements identified in LRB Table 2-1 (see Appendix A). Table 2-1 lays out a modular approach to the safety analysis review process. This SER and three supplements will be issued covering distinct chapters of the ABWR SSAR. In cases where early or late resolution of the review process is encountered, the individual chapters discussed in the SER and its supplements may differ slightly from those identified in Table 2-1.

1.2 General Description

To be provided in a supplement.

1.3 Comparison With Similar Facility Designs

To be provided in a supplement.

1.4 Identification of Agents and Contractors

To be provided in a supplement.

1.5 Summary of Principal Review Matters

To be provided in a supplement.

1.6 Modifications to the ABWR During the Course of the NRC Review

To be provided in a supplement.

1.7 Unresolved Safety Issues

The staff continuously evaluates the safety requirements used in its review against new information as it becomes available. The Commission's Severe Accident Policy Statement and 10 CFR Part 52 require that all new power plant designs address all unresolved safety issues (USIs) and all medium-and high-priority generic safety issues (GSIs). NUREG-1197, "Advanced Light Water Reactor Program," December 1986, presents these issues and their status as of July 1, 1986. GE is to identify and address issues that are applicable to the ABWR design. These issues will include both applicable issues identified in NUREG-1197 and any new generic issues raised up to the time of the issuance of the FDA. It is the intention of the staff that there will be no open items regarding the resolution of USIs or GSIs or other plant features for the ABWR when the FDA decision is made. The resolution of these issue will be discussed in Appendix C to later supplement.

1.8 Outstanding Issues

Certain outstanding regulatory review issues regarding the ABWR had not been resolved with GE when this report was issued. Many of these issues are in the unresolved category because the staff has not completed its review of information provided in the SSAR; other issues require additional information to be provided by GE. The staff will discuss the resolution of each of these items in a supplement to this report. These items and the sections (as indicated by the numbers in parentheses) in this SER or its supplements where they are or will be discussed as.

(1) site characteristics (2)
(2) design criteria for structures, systems, and components (3)
(3) fuel system design (4.2)
(4) nuclear design (4.3)
(5) thermal-hydraulic design (4.4)
(6) standby liquid control system reliability analysis (4.6)
(7) final report on fine motion control rod drive system in-plant test program (4.6)
(8) ASME Code Cases N-433 and N-451 (5.2.1.2)
(9) TMI-2 Action items related to safety/relief valves (5.2.2)
(10) inservice inspection and testing (5.2.4)
(11) preservice inspection (5.2.4)

(12) cleaning of stainless steel components (5.3.1)
(13) TMI-2 Action items related to emergency core cooling systems (5.4.6)
(14) containment systems (6.2)
(15) containment leak testing (6.2.4)
(16) control room habitability (6.4)
(17) atmosphere cleanup systems (6.5)
(18) main steam isolation valve leakage control (6.7)
(19) instrumentation and controls (7)
(20) electrical power systems (8)
(21) auxiliary systems (9)
(22) steam and power conversion system (10)
(23) radioactive waste management (11)
(24) radiation protection (12)
(25) conduct of operations (13)
(26) initial test program (14)
(27) transient and accident analysis (15)
(28) technical specifications (16)
(29) control room design review (18)
(30) severe accident design considerations (19)

1.9 Confirmatory Issues

At this point in the review, the following items have essentially been resolved to the staff's satisfaction. For these items, GE has already provided or has committed to provide the confirmatory information in the near future. If the staff's review of the information does not confirm its preliminary conclusions, that item will be treated as open and the staff will report its resolution in a supplement to this report.

(1) neutron fluence (5.3.2)
(2) reactor water cleanup system temperature capability (5.4.8)

1.10 Interface Information

The ABWR is a standard plant design based on the reference system concept. Although the ABWR is a complete plant design, certain features, such as site-dependent systems, may be outside the ABWR design scope. The design scope will

be defined in Section 1.2 of a supplement. The design scope is presently discussed in SSAR Section 1.2. The SSAR will also define the interface requirements imposed on an individual applicant who references the design so as to provide compatible design features that will ensure the applicability, functional performance, and safe operation of the ABWR systems.

Some of the interface requirements identified by the staff, with references to sections of this report, are the following. For a complete list of interface requirements, see the ABWR SSAR, Section 1.9.

(1) Fracture toughness (5.3.1)
(2) Steam isolation valve testing (5.4.6)

2 SITE CHARACTERISTICS

(This section will be provided in a supplement to this SER.)

3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS, AND COMPONENTS

(This section will be provided in a supplement to this SER.)

4 REACTOR

4.1 General

(To be provided in a supplement)

4.2 Fuel System Design

(To be provided in a supplement)

4.3 Nuclear Design

(To be provided in a supplement)

4.4 Thermal-Hydraulic Design

(To be provided in a supplement)

4.5 Reactor Materials

The acceptance criteria used as the basis for the staff's evaluation of reactor materials are found in Standard Review Plan (SRP) Section 4.5 (NUREG-0800).

4.5.1 Control Rod Drive System Structural Materials

The applicant has committed to the following: The properties of the materials selected for the control rod drive mechanism will be equivalent to those given in Appendix I to Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code); Parts A, B, and C of Section II of the ASME Code; and Regulatory Guide 1.85 "Code Case Applicability - ASME Section III Materials," except that cold-worked austenitic stainless steels shall have a 0.2-percent offset yield strength no greater than 90,000 pounds per square inch. All materials for use in this system will be selected for their

compatibility with the reactor coolant as described in Articles NB-2160 and NB-3120 of the ASME Code. The materials selected will be resistant to stress corrosion in a BWR environment.

The controls imposed on the austenitic stainless steel of the control rod drive mechanism conform to the recommendations of Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," and Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel."

All materials selected for application in control rod drive mechanism components are or will be in conformance with the applicable code case listed in Regulatory Guide 1.85. Fabrication and heat treatment practices that will be performed in accordance with these recommendations provide added assurance that stress corrosion cracking will not occur during the design life of the components. Both martensitic and precipitation-hardening stainless steels will be given tempering or aging treatments in accordance with staff positions. Cleaning and cleanliness controls are in accordance with American National Standards Institute Standard N45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants," and Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

The staff concludes that the structural materials selected for the control rod drive mechanism are acceptable and meet the requirements of General Design Criteria (GDC) 1, 14, and 26 of Appendix A to 10 CFR Part 50 and of 10 CFR 50.55a.

4.5.2 Reactor Internal Materials

The applicant has met the requirements of GDC 1 and 10 CFR 50.55a with respect to ensuring that the design, fabrication, and testing of the materials to be used in the reactor internals and core support structures are of high-quality standards and adequate for ensuring structural integrity. The controls imposed on components to be constructed of austenitic stainless steel satisfy the recommendations of Regulatory Guides 1.31 and 1.44.

The materials to be used for construction of the components of the reactor internals and the core support structures have been identified by specification and conform with the requirements of Article NG-2000 of Section III and Parts A, B, and C of Section II of the ASME Code. In addition, the applicant has met the guidelines of Regulatory Code 1.85 by specifying construction materials that are approved for use by the ASME Code cases. As proven by extensive tests and satisfactory performance, the specified materials are compatible with the BWR environment.

The controls imposed on the reactor coolant chemistry provide reasonable assurance that the reactor internals and the core support structures will be adequately protected during operation from conditions that could lead to stress corrosion of the materials and loss of component structural integrity.

The material selection, fabrication practices, examination and testing procedures, and control practices to be performed in accordance with these recommendations provide reasonable assurance that the materials to be used for the reactor internals and core support structures will be in a metallurgical condition to preclude inservice deterioration. Conformance with the requirements of the ASME Code and the recommendations of the regulatory guide constitutes an acceptable basis for meeting, in part, the requirements of GDC 1 and 10 CFR 50.55a.

The staff concludes that the materials to be used for the construction of the reactor internals and core support structures are acceptable and meet the requirements of GDC 1 and 10 CFR 50.55a.

4.6 Functional Design of Fine Motion Control Rod Drive System

The fine motion control rod drive system (FMCRD) was reviewed in accordance with SRP Section 4.6. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for the staff's evaluation of the control rod drive system with respect to the applicable regulations of 10 CFR Part 50.

The ABWR design incorporates electric-hydraulic fine motion control rod drives, which will provide electric fine rod motion during normal operation and hydraulic pressure for scram insertion. Fine motion (18 millimeter) during normal operation will be provided by a ball screw nut, and the rods will be shaft driven by the electric stepper motor. In response to a scram signal, the control rods will be inserted hydraulically via the stored energy in the scram accumulator this action is similar to that of current operating BWR control rod drives. A scram signal also will be given simultaneously to insert the FMCRD electrically via the FMCRD motor drive. The use of hydraulic and electric methods of scramming provides a high degree of assurance that rods will be inserted on demand.

The FMCRDS and the recirculation flow control system are designed to control reactivity during power operation. In the event of fast transients, reactivity will be controlled by automatic rod insertion. During an anticipated transient without scram (ATWS), the internal recirculation pumps will be tripped automatically. If the reactor cannot be shut down with the control rods, the operator will be able to actuate the standby liquid control system, which will pump a solution of sodium pentaborate into the primary system. The evaluation of the functional design of the standby liquid control system will be found in Section (9.3.5) of a supplement to this SER.

Reactivity in the core will be controlled by the FMCRD by moving control rods interspersed throughout the core. These rods control the reactor's overall power level and provide the principal means of quickly and safely shutting down the reactor.

A single hydraulic control unit (HCU) will power the scram action of two FMCRD. A supply pump (with a spare pump on standby) will provide the HCUs with water from the condensate storage tank for supplying contol rod drive (CRD) purge water and purge water to reactor internal pumps and the reactor water cleanup RWCU pumps. The pump also will provide water to a scram accumulator in each HCU to maintain the desired water inventory. When necessary, the accumulator will force water into the drive system to scram the control rods connected to that HCU; the volume of water in the scram accumulator will be sufficient to scram the rod. A single failure in an HCU would result in the failure of two rods

only. These would not be adjacent rods and would be separated sufficiently so that the reactivity effect would essentially be the same as for the failure of one rod. There is adequate shutdown margin with a single failure, even though the HCU will be shared by two drives.

The FMCRDS has been designed to permit periodic functional testing during power operation. The capability to independently test the individual scram channels and the motion of individual control rods. The FMCRDS is designed so that failure of all electrical power or instrument air will cause the control rods to scram, thereby protecting the reactor. On the basis of the above, the staff concludes that the requirements of GDC 23 are satisfied.

Preoperational tests of the control rod drive hydraulic system will be conducted to determine capability of the system. Startup tests will be conducted over the range of temperatures and pressures from shutdown to operating conditions in order to determine compliance with applicable technical specifications. Each rod that is partially or fully withdrawn during operation will be exercised one notch at least once each week. Control rods will be tested for compliance with scram time criteria, from the fully withdrawn position, after each refueling shutdown.

The FMCRDS is designed to control reactivity under normal operating conditions and during anticipated operational occurrences. This capability is demonstrated by the safety analyses discussed in SSAR Chapter 15 (including the effects of stuck rods). This system is also capable of holding the core subcritical under cold shutdown conditions. The recirculation flow control system is capable of accommodating reactivity changes during normal operating conditions. The standby liquid control system is capable of bringing the reactor subcritical under cold shutdown conditions in the event the control rods cannot be inserted. These systems, taken together, satisfy the requirements of GDC 26, 27, and 29.

The control rod system design incorporates appropriate limits on the potential amount and rate of reactivity increase. Control rod withdrawal sequences and patterns will be selected to achieve optimum core performance and low individual rod worths. The rod pattern control system (RPCS) will reduce the chances of withdrawal other than by the preselected rod withdrawal pattern. The RPCS

function assists the operator with an effective backup control rod monitoring routine that enforces adherence to established startup, shutdown, and low-power operation control rod procedures.

A malfunction in the FMCRDS could result in a reactivity change. The applicant demonstrated in SSAR Chapter 15 that the FMCRDS will limit these postulated transients to within acceptable fuel response limits, as required by GDC 25.

The control rod mechanical design incorporates a brake system that will reduce the chances of rapid rod ejection. This engineered safeguard will protect against a high reactivity insertion rate from a potential control rod ejection. Normal rod movement and the rod withdrawal rate will be limited through the fine motion control motor. The ABWR control rod design does not include a velocity limiter.

An internal CRD housing support has been adopted to replace the support structure of beams, hanger rods, grids, and support bars used in current BWR designs. This system utilizes the outer tube of the drive to provide support. This tube which is welded to the drive middle flange, attaches by a bayonet lock to the guide tube base. The guide tube being supported by the housing extension prevents downward movement of the drive in the event of housing failure. The CRD housing support is designed to prevent ejection of a CRD and attached control rod.

The fine motion control rod drive is designed to detect separation of the control rod from the drive mechanism. Two redundant and separate Class 1E switches are provided to detect the separation of either the control rod from the hollow piston or the hollow piston from the ball nut. Actuation of either of these switches will cause an immediate rod block and initiate an alarm in the control room, thereby reducing the chances of a rod drop accident from occurring. The ABWR control rod design does not include a velocity limiter because of the design features described.

The design features of the reactivity control system that limit the potential amount and rate of reactivity increase ensure that GDC 28 is satisfied for postulated reactivity accidents.

The concerns discussed in NUREG-0803, "Safety Concerns Associated With a Pipe Break in the BWR Scram System," are not applicable for the ABWR. There is no scram discharge volume piping the ABWR design. The scram discharge volume will be routed to the reactor pressure vessel.

The staff is in the process of evaluating the ABWR design against the requirements of the ATWS rule 10 CFR 50.62. Since the ABWR standby liquid control system pump will be started manually rather than automatically as required by the ATWS rule, detailed technical justification should be submitted for staff review. This technical justification should include a detailed reliability analysis to support this deviation. The analysis should include the control room displays, emergency procedures, and the time required for operator action. As part of this analysis, the 86-gallon per-minute boron equivalent requirement should be demonstrated by the ABWR design. Also, GE should provide a quantitative analysis to demonstrate that the new electric scram system and the alternate rod insertion system reduce the probability of an ATWS.

In response to Question 440.8 in the SSAR, GE stated that the final report on the FMCRD in-plant test program, which will include the LaSalle test results, will be issued in September 1989. The staff will review the results and report the findings in a supplement to this report.

On the basis of its review, the staff concludes, subject to the open issues discussed above, that the functional design of the reactivity control system conforms to the requirements of GDC 23, 25, 26, 27, 28, and 29 with respect to demonstrating the ability to reliably control reactivity changes under normal operation, anticipated operational occurrences, and accident conditions including single failures, and is, therefore, acceptable. The functional design of the reactivity control system conforms to the applicable acceptance criteria of SRP Section 4.6.

5 REACTOR COOLANT SYSTEM

5.1 Summary Description

The reactor coolant system (RCS) as designed includes those systems and components that will contain or transport fluids coming from or going to the reactor core. These systems form a major portion of the reactor coolant pressure boundary (RCPB). This section provides information on the reactor coolant system and pressure-containing appendages out to and including isolation valving. This group of components is defined as the RCPB.

The RCPB includes all pressure-containing components, such as pressure vessels, piping, pumps, and valves, that are

(1) part of the RCS, or

(2) connected to the RCS up to and including any and all of the following:

   (a) the outermost containment isolation valve in piping that penetrates the primary reactor containment,

   (b) the second of the two valves normally closed during normal reactor operation in system piping that does not penetrate primary reactor containment

   (c) the RCS safety/relief valve piping

This section also deals with various subsystems to the RCPB that are closely allied to it.

The nuclear system pressure relief system will protect the RCPB from damage due to overpressure. To protect against overpressure, pressure-operated relief valves are provided that will discharge steam from the nuclear system to the suppression pool. The pressure relief system also will act to automatically depressurize the nuclear system in the event of a loss-of-coolant accident (LOCA) during which the feedwater, reactor core isolation cooling, and high-pressure

core flooder systems fail to maintain reactor vessel water level. Depressurization of the nuclear system will allow the low-pressure flooder systems to supply enough cooling water to adequately cool the fuel.

The major safety consideration for the reactor vessel is the ability of the vessel to function as a radioactive material barrier. Various combinations of loading are considered in the vessel design. The vessel meets the requirements of applicable codes and criteria. The possibility of brittle facture was considered, and suitable design, material selection, material surveillance activity, and operational limits were established to avoid conditions where brittle fracture was possible.

The reactor recirculation system (RRS) will provide coolant flow through the core. Adjustment of the core coolant flow rate changes reactor power output, thus providing a means of following plant load demand without adjusting control rods. The RRS is designed to provide a slow coastdown of flow so that fuel thermal limits cannot be exceeded as a result of recirculation system malfunctions. The reactor recirculation pumps will be located inside the reactor vessel, thus eliminating large piping connections to the reactor vessel below the core and also eliminating the reactor recirculation system piping.

The main steamline flow restrictors of the venturi type will be installed in each main steam nozzle on the reactor vessel inside the primary containment. The restrictors are designed to limit the loss of coolant resulting from a main steamline break inside or outside the primary containment. The coolant loss is limited so that reactor vessel water level will remain above the top of the core during the time required for the main steamline isolation valves to close. This action protects the fuel barrier.

Two isolation valves will be installed on each main steamline. One will be located inside and the other outside the primary containment. If a main steamline break should occur inside the containment, closure of the isolation valve outside the primary containment will seal the primary containment itself. The main steamline isolation valves will automatically isolate the RCPB if a pipe break should occur outside the containment. This action will limit the loss of coolant and the release of radioactive materials from the nuclear system.

The reactor core isolation cooling system will provide makeup water to the core during a reactor shutdown in which feedwater flow is not available. The system will be started automatically on receipt of a low reactor water level signal or manually by the operator. Water will be pumped to the core by a turbine pump driven by reactor steam.

The residual heat removal (RHR) system will include a number of pumps and heat exchangers that can be used to cool the nuclear system in a variety of situations. During normal shutdown and reactor servicing, the RHR system will remove residual and decay heat. The RHR system will allow decay heat to be removed whenever the main heat sink (main condenser) is not available (i.e., hot standby). One mode of RHR operation allows the removal of heat from the primary containment following a LOCA. Another operational mode of the RHR system is low-pressure flooding. The low-pressure flooder is an engineered safety feature for use during a postulated LOCA.

The reactor water cleanup system will recirculate a portion of reactor coolant through a filter demineralizer to remove particulate and dissolved impurities with their associated corrosion and fission products from the reactor coolant. It also will remove excess coolant from the reactor system under controlled conditions.

5.2 Compliance With Code and Code Cases

This section discusses and confirms the staff's review of measures to be used to provide and maintain the integrity of the reactor coolant pressure boundary and other components that are important to safety for the plant design lifetime.

5.2.1 Compliance With 10 CFR 50.55a

10 CFR 50.55a, "Codes and Standards," requires the following for components important to safety:

(1) Components in the reactor coolant pressure boundary must meet the requirements for Class 1 (Quality Group A) components in ASME Code, Section III, except for those components that meet the exclusion requirements of 10 CFR 50.55a(c)(2).

2. Components classified as Quality Groups B and C must meet the requirements for Class 2 and 3 components, respectively, in ASME Code, Section III.

The pressure-retaining components of the RCPB as defined by 10 CFR 50.55a have been properly classified in Table 3.2-1 of the ABWR Standard Safety Analysis Report (SSAR) as ASME Code, Section III, Class 1 components. These Section III, Class 1 components are designated Quality Group A in conformance with Regulatory Guide (RG) 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants." The staff reviewed the Quality Group A RCPB components in accordance with Standard Review Plan (SRP) Section 5.2.1.1 (NUREG-0800), and the results of this review are discussed later in this section of the SER.

In addition to the Quality Group A components of the RCPB, certain lines that will perform a safety function and that meet the exclusion requirements of 10 CFR 50.55a(c)(2) are classified Quality Group B in accordance with Position C.1 of RG 1.26 and will be constructed as ASME Code, Section III, Class 2 components. The staff review of these components and other pressure-retaining components that will be constructed to ASME Code, Section III, Class 2 and Class 3 will be discussed in Section 3.2.2 of a supplement to this SER.

The review procedures of SRP Section 5.2.1.1 recommend that safety analysis reports for both construction permits and operating licenses contain a table identifying the ASME component code, code edition, and applicable addenda for all ASME Codes, Section III, Class 1 and 2 pressure vessel components, piping, pumps, and valves in the reactor coolant pressure boundary. This table is reviewed by the staff for compliance with 10 CFR 50.55a. Section 5.2.1.1 of the SSAR contains a commitment that the ASME Code edition, applicable addenda and component dates will be in accordance with 10 CFR 50.55a. The ASME code editions, addenda, and dates are identified in SSAR Tables 1.8-21 and 3.2-3.

On the basis of its review of Sections 3.2 and Section 5.2.1.1 of the SSAR as discussed above, the staff concludes that the construction of components of the RCPB in the ABWR nuclear island will conform with the appropriate ASME Code editions and addenda and the Commission's regulations and therefore provides assurance that component quality will be commensurate with the importance of the safety function of the RCPB. This constitutes an acceptable basis for satisfying GDC 1 and is, therefore, acceptable.

5.2.1.2 Applicable Code Cases

Table 5.2-1 in the SSAR identifies specific ASME Code cases that will be applied in the construction of pressure-retaining ASME Code, Section III, Class 1, 2, and 3 components. The staff's review of this table is based on the guidelines in Regulatory Guide 1.84, "Design and Fabrication Code Case Acceptability - ASME Section III, Division 1," and RG 1.85, "Materials Code Case Acceptability - ASME Section III, Division 1." The Code cases listed below are in Table 5.2-1, but have not yet been endorsed by the staff in either RG 1.84 or 1.85.

(1) Code Case N-433, "Non-Threaded Fasteners for Section III, Division 1, Class 1, 2, and 3 Component and Piping Supports"

(2) Code Case N-451, "Alternative Rules for Analysis of Piping Under Seismic Loading - Section III, Division 1, Class 1"

Code Case N-414, "Tack Welds for Class 1, 2, and 3 and MC Components and Piping Supports," was unconditionally accepted by the staff in RG 1.84, Revision 25, dated May 1988. The staff, however, is currently reassessing its review of this Code case and, if applicable, will report the results of its review in a future revision of RG 1.84.

With the exception of Code Cases N-433 and N-451, the staff concludes that the Code cases in Table 5.2-1 of the SSAR meet the guidelines of RG 1.84 and 1.85 and are acceptable. Compliance with the requirements of these Code cases will result in a component quality level that is commensurate with the importance of the safety function of these components and constitutes the basis for satisfying the requirements of 10 CFR Part 50, Appendix A, General Design Criterion 1 and is, therefore, acceptable.

5.2.2 Overpressure Protection

The staff performed its evaluation of overpressure protection in conformance with the review guidelines and acceptance criteria of SRP Section 5.2.2, which states that the acceptance criteria are also based on General Design Criterion (GDC) 31 of 10 CFR Part 50, Appendix A, as it relates to the fracture behavior of the reactor coolant pressure boundary (RCPB). This review area is addressed in Section 5.3.1 of this SER.

The RCPB is provided with a pressure relief system to

(1) prevent the pressure within the RCPB from rising beyond 110 percent of the design value

(2) provide automatic depressurization in the event of small breaks in the nuclear system occurring together with failure of the high-pressure core flooder system so that the low-pressure flooder systems can operate to protect the fuel barrier

The relief system must permit verification of its operability and withstand adverse combinations of loadings and forces resulting from normal, upset, emergency, and faulted conditions.

Overpressurization protection in the ABWR will be accomplished through the use of 18 safety/relief valves (SRVs). The 18 SRVs are divided into 5 groups according to nominal pressure setpoints and will be mounted on the four main steamlines between the reactor vessel and the first isolation valve inside the drywell. The SRVs will discharge through piping to the suppression pool. The ABWR pressure relief system design is similar to that of other BWR Class 4, 5, and 6 plants.

The SRVs are classified as Quality Group A and seismic Category I, as shown in Table 3.2-1 of the SSAR. The SRVs will be designed to meet Regulatory Guides 1.26 and 1.29, "Seismic Design Classification."

The nominal pressure setpoints of the SRVs are distributed in five valve groups with a minimum setpoint of 1150 psig and a maximum of 1190 psig in the safety mode of operation. The nominal pressure setpoints of the SRVs in the relief mode of operation are 1090 psig and 1140 psig. The SRVs will be able to operate in the power-actuated mode by remote-manual controls from the main control room. The effects of flow-induced SRV discharge line backpressure on the performance of the SRV are addressed by sizing the line to ensure that the steady-state backpressure does not exceed 40 percent of the SRV inlet pressure. This sizing criterion controls the effective backpressure buildup and maintains the required force balance needed to keep the SRV open and to permit proper blowdown.

Before the valves will be installed, the SRV manufacturer will test them hydrostatically for response set pressure and seat leakage to certify that design performance requirements have been met. During the preoperational test program, specified manual and automatic actuation will be verified, as recommended in RG 1.68, "Initial Test Programs for Water-Cooled Reactor Power Plants."

GDC 15 defines the basis for overpressurization protection in a nuclear reactor. It requires that the RCPB design conditions not be exceeded during any condition of normal operation, including anticipated operational occurrences.

To satisfy this criterion, the overpressurization protection system for the ABWR has been designed in compliance with ASME Code, Section III, which requires that the maximum pressure reached during the most severe pressure transient be less than 110 percent of the design pressure. For the ABWR that pressure limit is 1375 pounds per square inch gauge (psig).

GE has analyzed a series of transients that would be expected to require pressure relief actuation to prevent overpressurization. The analysis was performed using the computer-simulated model ODYNA. ODYNA is the improved version of ODYN. ODYN is described in General Electric Topical Report NEDO-24154. The ODYN code has been reviewed by the staff and found acceptable. (Safety Evaluation for the General Electric Topical Report

Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors, NEDO-24154 and NEDO-24154-P Volumes I, II, III, June 1980). The results of these analyses demonstrate that the maximum pressure will remain below the 1375 psig limit. The SRVs are direct-acting oevices that are assumed to actuate at the spring setpoint during the safety mode of operation. For the most severe transient, closure of all main steam isolation valves (MSIV) with a high neutron flux scram, the maximum vessel bottom pressure is calculated to be 1274 psig when all 18 SRVs are assumed to operate in the safety mode. For the analysis, it was assumed that the plant was operating at 102.7 percent of rated steam flow (17.29×10^6 lb/hr) and a vessel dome pressure of 1040 psig. GE considered the effects of recirculation pump trips in the analysis.

GE has based the sizing of the SRVs on the initiation of a reactor scram by the high-neutron flux scram, which is the second safety-grade scram signal from the reactor protection system following MSIV closure. The staff believes that the qualification and redundancy of reactor protection system equipment, coupled with the fact that the reactor vessel pressure is limited to less than 110 percent of design pressure, provide adequate assurance that the reactor vessel integrity will be maintained for the limiting transient event.

TMI-2 Action Plan (NUREG-0737) items related to SRVs - II.B.1, II.D.1, - II.D.3, II.K.3.16, and II.K.3.28, will be discussed in a supplement to this SER.

The staff concludes that the pressure relief system, in conjunction with the reactor protection system, will provide adequate protection against overpressurization of the RCPB. The staff concludes that the overpressurization system is acceptable and meets the relevant requirements of GDC 15, "Reactor Coolant System Design."

5.2.3 Reactor Coolant Pressure Boundary Materials

The materials to be used for construction of components of the RCPB have been identified by specification and conform with the requirements of Section III of the ASME Code and NUREG-0313. The applicant, however, must use Revision 2 not Revision 1 of NUREG-0313. Compliance with the above Code provisions for material specifications satisfies the quality standards of GDC 1 and 10 CFR 50.55a.

The materials to be used for the construction of the RCPB that will be exposed to the reactor coolant have been identified, and all of the materials are compatible with the primary coolant water, which will be chemically controlled in accordance with appropriate technical specifications. This compatibility has been proven by extensive testing and satisfactory performance. This includes conformance with the recommendations of Regulatory Guide 1.44, "Control of Sensitized Stainless Steel," and the requirements of NUREG-0313. General corrosion of all materials, except unclad carbon and low alloy steel, will be negligible. For these materials, conservative corrosion allowances have been provided for all exposed surfaces in accordance with the requirements of ASME Code, Section III. The above evidence of compatibility with the coolant and compliance with the Code provisions satisfy the requirements of GDC 4 relative to compatibility of components with environmental conditions.

The materials to be used for the construction of the RCPB are compatible with the thermal insulation to be used in these areas and conform with the recommendations of Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steels." Conformance with the above recommendations satisfies the requirements of GDC 14 and 31 relative to the prevention of failure of the RCPB.

The ferritic steel tubular products and the tubular products fabricated from austenitic stainless steel have been found to be acceptable by nondestructive examinations in accordance with the provisions of ASME Code, Section III, and Regulatory Guide 1.66. Compliance with these Code requirements satisfies the quality standards of GDC 1 and 30 and 10 CFR 50.55a.

The fracture toughness tests required by the ASME Code, augmented by Appendix G to 10 CFR Part 50, provide reasonable assurance that adequate safety margins against nonductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the RCPB. The use of Appendix G to ASME Code, Section III, and the results of fracture toughness tests performed in accordance with the Code and NRC regulations in establishing safe operating procedures will provide adequate safety margins during operating, testing, maintenance, and postulated accident conditions. Compliance with these Code provisions and NRC regulations satisfies the requirements of GDC 31 and 10 CFR 50.55a regarding the prevention of fracture of the RCPB.

The controls imposed on preheat temperatures for welding of ferritic steels conforms with the recommendations of Regulatory Guide 1.50, "Control of Preheat Temperature for Welding Low Alloy Steels." These controls provide reasonable assurance that cracking of components made from low-alloy steels will not occur during fabrication and minimize the possibility of subsequent cracking due to residual stresses retained in the weldment. These controls satisfy the quality standards of GDC 1 and 30 and 10 CFR 50.55a.

Controls imposed on electroslag welding of ferritic steels are not necessary because electroslag welding will not be used for RCPB components.

The controls imposed on the welding of ferritic steels under conditions of limited accessibility are in accordance with the recommendations of Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility," and provide assurance that proper requalification of welders will be required in accordance with the welding conditions. These controls also satisfy the quality standards of GDC 1 and 50 and 10 CFR 50.55a.

The controls imposed on weld cladding of low-alloy steel components by austenitic stainless steel are in accordance with the recommendations of Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components." These controls provide assurance that practices that could result in underclad cracking will be restricted. They also satisfy the quality standards of GDC 1 and 30 and 10 CFR 50.55a.

The controls to prevent stress corrosion cracking in RCPB components constructed of austenitic stainless steel limit yield strength of cold-worked austenitic stainless steel to 90,000 psi maximum and conform to the recommendations of Regulatory Guides 1.44, "Control of the Use of Sensitized Stainless Steel," and 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Plants." The controls to be followed during material selection, fabrication, examination, and protection, in accordance with these recommendations, in order to prevent excessive yield strength, sensitization, and contamination, provide reasonable assurance that the RCPB components of austenitic stainless steel will be in a metallurgical condition

that minimizes susceptibility to stress corrosion cracking during service. These controls meet the requirements of GDC 4 relative to the compatibility of components with environmental conditions and of GDC 14 relative to the prevention of leakage and failure of the RCPB.

The staff concludes that the reactor coolant pressure boundary materials are acceptable and meet the requirements of GDC 1, 4, 14, 30, and 31 of Appendix A of 10 CFR Part 50; Appendices B and G to 10 CFR Part 50; and 10 CFR 50.55a.

5.2.4 Inservice Inspection and Testing of Reactor Coolant Pressure Boundary

Section 5.2.4 of the SSAR describes provides a generic inservice inspection (ISI) program; however, the ISI program for a specific plant needs more detailed discussion on all phases of the inspection, such as the specific components and systems that will be inspected including drawings, the methodology of the inspection, etc. GE did not discuss the preservice inspection (PSI) in this section of the SSAR. The staff review was conducted in accordance with SRP Section 5.2.4.

5.2.4.1 Compliance With Standard Review Plan

The review according to SRP Section 5.2.4, Paragraph II.3, "Examination Categories and Methods," has not been completed because the applicant has not submitted a PSI program.

The review according to SRP Section 5.2.4, Paragraph II.4, "Inspection Intervals," has not been completed because the inspection interval of the PSI program was not addressed in the SSAR.

The review according to SRP Section 5.2.4, Paragraph II.5, "Evaluation of Examination Results," has been done. The applicant committed to evaluate the results in accordance with ASME Code, Section XI, Article IWB-3000, "Acceptance Standards for Flaw Indications," and repairs will be based on the requirements of Articles IWA-4000 and IWB-4000, "Repair Procedures." However, ongoing NRC generic activities and research projects indicate that the acceptable-size flaws

specified in Article IWB-3000 may not always be detected using the currently specified ASME Code procedures. For example, the acceptable-size flaws were not detected in all cases by using the ASME Code procedures specified for volumetric examination of the reactor vessel, bolts and studs, and piping. The staff will continue to evaluate the development of improved procedures and will require that these improved procedures be made a part of the inservice examination requirements.

The review according to SRP Section 5.2.4, Paragraph II.7, "Acceptance Criteria, Code Exemptions," will be performed when a complete PSI program is submitted by GE. Although exemptions from Code examinations are permitted, the PSI program must list the exemptions taken and the criteria in accordance with the Code.

The review according to SRP Section 5.2.4, Paragraph II.8, "Acceptance Criteria, Relief Requests," has not been done because the applicant has not identified limitations to examination. Specific areas where ASME Code examination requirements cannot be met are normally identified during the performance of the PSI. GE should identify all plant-specific areas where ASME Code, Section XI requirements cannot be met and provide a supporting technical justification for relief.

5.2.4.2. Examination Requirements

General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary," of Appendix A to 10 CFR Part 50 requires that components that are part of the reactor coolant pressure boundary be designed to permit periodic examination and testing of important areas and features to assess their structural and leaktight integrity. To ensure that no deleterious defects develop during service, selected welds and weld heat-affected zones are to be examined periodically.

The design of the ASME Code, Class 1 and 2 components of the reactor coolant pressure boundary incorporates provisions for access for inservice examinations, as required by Article IWA-1500 of Section XI of the ASME Code. 10 CFR 50.55a(g) defines the detailed requirements for the preservice and inservice programs for light-water-cooled nuclear power facility components. On the basis of the construction permits issued by the NRC on or after July 1, 1974, this section of the regulations requires that a preservice inspection program be developed and

implemented using the edition and addenda of Section XI of the ASME Code applied to the construction of the particular components. The applicable ASME Code editions and addenda are identified in SSAR Table 1.8-21.

5.2.4.3 Evaluation of Compliance With 10 CFR 50.55a(g)

The review of compliance with 10 CFR 50.55a(g) is incomplete because the applicant has not specifically discussed compliance with 10 CFR 50.55a(g) in SSAR Section 5.2.4.

The specific areas where the applicable Code requirements cannot be met can only be identified after the examinations are performed. GE should commit to identify all plant-specific areas where the Code requirements cannot be met and provide a supporting technical justification for relief. The plant-specific SER will be completed after the applicant referencing the ABWR design performs the following:

(1) Dockets a complete and acceptable PSI program plan. The PSI program should include reference to the ASME Code, Section XI edition and addenda that will be used for the selection of components for examination, lists of the components subject to examination, a description of the components exempt from examination by the applicable code, and the examination isometric drawings.

(2) Submits plans for preservice examination of the reactor pressure vessel welds to address the degree of compliance with Regulatory Guide 1.150, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations."

(3) Discusses the near-surface examination and resolution with regard to detecting service-induced flaws and the use of electronic gating as related to the volume of material near the surface that is not being examined. Discusses how the internal surfaces (e.g., inner radius of a pipe section and reactor vessel internals) will be examined.

(4) Dockets an acceptable resolution of the information requested regarding the PSI/ISI program.

(5) Submits all relief requests, if needed, with a supporting technical justification.

The staff considers the review of the GE ABWR PSI and ISI programs an open issue subject to the resolution of the above items.

5.2.4.4 Conclusions

The conduct of periodic examinations and hydrostatic testing of pressure-retaining components of the reactor coolant pressure boundary, in accordance with the requirements of Section XI of the ASME Code and 10 CFR Part 50, will provide reasonable assurance that structural degradation or loss of leaktight integrity occurring during service will be detected in time to permit corrective action before the safety functions of a component are compromised. Compliance with the preservice and inservice examinations required by the Code and 10 CFR Part 50 constitutes an acceptable basis for satisfying the inspection requirements of GDC 32.

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

The staff reviewed the RCPB leakage detection systems in accordance with SRP Section 5.2.5 (NUREG-0800) and performed an audit review of each of the areas listed in the "Areas of Review" portion of the SRP section according to the guidelines in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for the staff's evaluation of the RCPB leakage detection systems with respect to the applicable regulations of 10 CFR Part 50.

A limited amount of leakage is to be expected from components forming the RCPB. Means are provided for identifying this leakage in accordance with the requirements of GDC 30, "Quality of Reactor Coolant Pressure Boundary."

Leakage is classified into two types, identified and unidentified. Components such as valve stem packing, pump shaft seals, and flanges are not completely leaktight. Because some leakage is expected, it is considered identified leakage and is monitored separately from unidentified leakage (which may be symptomatic of an unexpected failure of the RCPB) in accordance with Position C.1 of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems." Sensitivity of detection methods (1 gpm) and leakage limits (ranging from 1 gpm to 5 gpm depending on the specific plant that references the ABWR design) for unidentified leakage are selected to detect and correct potential through-wall flaws (cracks) in the RCPB before such cracks can grow sufficiently to threaten the safety of the plant.

Reactor coolant leakage for the ABWR involves the following: (1) leakage within the drywell, (2) leakage external to the drywell (i.e., in the equipment areas in the reactor building, in the main steam tunnel, and in the turbine building) and (3) intersystem leakage. These are discussed below.

Leakage Within the Drywell

Identified Leakage

Within the drywell, the drywell equipment drain sump will collects leakage from the reactor vessel head flange inner seal, stem inner packing for large (greater than 2 inches) remote power operated valves, and other known leakage sources. The sump will be equipped with two pumps, two timers, and instrumentation for control room monitoring. The sump instrumentation and timers will monitor identified leakage by measuring the sump level rate of change and the sump fillup and pumpout times. Control room indication and alarm capabilities will be provided. The sump level monitoring instrument and the fillup and/or pumpout timer will activate an alarm in the control room when the total leak rate reaches 25 gpm.

Different instrumentation and parameters will be monitor leakage from individual sources. The head flange inner seal leakage will be monitored by means of a seal drain line pressure instrument. Safety/relief valve (SRV) leakage will be monitored by a temperature sensor provided on each SRV discharge line. These monitors will continuously indicate and/or record leakage in the control room. In addition, the monitors will trip and activate an alarm in the control room if leakage from the monitored components is detected. The stems for large power operated valves will be equipped with drain lines to the sump. A remote operated solenoid valve is installed on each drain line can be used during plant operation, in conjunction with additional sump instrumentation, to identify leaking valve packing and to isolate leakage flow from the valve stem inner seal.

Unidentified Leakage

Within the drywell, the drywell floor drain sump will collect unidentified leakage from sources such as control rod drives, valve flanges, closed cooling water for reactor services (e.g., reactor internal pump (RIP) motor cooling), condensate from the drywell atmosphere coolers, and other leakage not collected in the drywell equipment drain sump. The sump will be equipped with two pumps, two timers, and instrumentation for control room monitoring. Sump fillup and pumpout times will be monitored. The instrumentation will activate an alarm in the control room when preset limits are reached. Continuous indication of sump level rate of change will be provided in the control room. Other primary detection methods for small unidentified leaks include increases in condensate flow rate increases from the drywell air coolers and increases in radioactivity count level for noble gases, iodines, and particulates in the drywell atmosphere. The flow rate for condensate, which results from condensation of thermally hot leakage inside the drywell, will be monitored by flow instrumentation provided on the common drain line for the condensate from all the coolers. The count levels will be monitored by drywell radiation monitors. These variables will be continuously indicated and/or recorded in the control room. The monitors will activate alarms in the control room when their preset limits are reached. The sensitivity and response time for all these primary detection systems are 1 gpm

or its equivalent in less than 1 hour, thus satisfying Positions C.2 and C.5 of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection System." Secondary detection methods include monitoring the drywell temperature and pressure for gross unidentified leakage. These variables will be recorded in the control room and will activate control room alarms when they are high. Excessive leakage inside the drywell (e.g., during a loss-of-coolant accident) will be detected by high drywell pressure, low reactor vessel water pressure, or high steamline flow (for breaks downstream of the flow elements). The instrumentation for these monitored variables will trip, activate alarms, and isolate the appropriate valves when their predetermined limits are exceeded. The primary and secondary detection methods described above meet Position C.3 of Regulatory Guide 1.45.

Leakage External to the Drywell

The areas designated as external to the drywell that will be monitored for reactor coolant leakage are (1) the equipment areas in the reactor building, (2) the main steam tunnel, and (3) the turbine building. The process piping for each monitored system will be located in separate compartments or rooms, where feasible, to facilitate the detection of the leak through area temperature indications. Each leakage detection system has the capability of detecting leak rates that are less than the established limits for identified and unidentified leakage from the RCPB. Many detection and monitoring systems will be used to detect leakage external to the drywell.

Equipment Areas in the Reactor Building

Leakage from unknown or unidentified sources, including shutdown cooling system piping, reactor water cleanup (RWCU) system piping, process instrumentation piping, and control rod drive (CRD) hydraulic control unit (HCU) piping, will be collected in several reactor building floor drain sumps. Identified leakge from known sources such as the following will be collected in reactor building equipment drain sumps:

- reactor core isolation cooling (RCIC), residual heat removal (RHR), and high-pressure core flooder pump shaft seal drains, pump seal vent, and suction vent discharges

- RHR heat exchanger drains

- valve stem packing drains via return from valve gland leakage treatment system drain lines

- process piping vent, leaktightness test, and sample drain lines

- HCU equipment drains

- CRD system pumps drain and vent lines and filter drains and vents

Pumps, timers, and instrumentation used for processing and monitoring the reactor building equipment and floor drain sumps will be the same as those provided for the drywell equipment and floor drain sumps. Sump levels and sump fillup and pumpout times will be monitored. Alarms will be initiated in the control room when the applicable setpoints are reached.

Besides the above parameters (i.e., sump levels and sump fillup and pumpout times), other parameters also will be used for detecting leakage. Equipment rooms that include RCIC, RHR, and RWCU system components will be monitored by dual-element thermocouples that sense high ambient temperature. These sensors will indicate reactor coolant leaks in these areas. The high-temperature condition will be indicated, alarmed, and recorded in the control room. In some cases, a high-temperature condition will provide isolation signals to close appropriate valves. These monitors are suitable for detecting leakage of 25 gpm or less into the monitored areas. In addition to area temperature monitoring, leakage external to the drywell originating from specific systems can be detected by monitoring other parameters such as low steam pressure and high steam flow in RCIC and main steamlines, high RCIC exhaust line diaphragm pressure, and high differential flow between suction and discharge lines of the RWCU system.

These detection systems will activate alarms in the control room and/or initiate closure of applicable valves when the preset limits for the corresponding parameters are reached.

In Main Steam Tunnel

Leakage in the main steam tunnel will be detected by monitoring area temperatures and radiation. Temperature in the main steamline area will be monitored by dual-element thermocouples located in the area of the main steam and RCIC pipelines. All temperature elements will be located and shielded so that they will be sensitive to ambient air temperature rather than the radiated heat from hot equipment. The monitors will alarm in the control room and provide isolation signals to appropriate control systems when setpoints are reached. These monitors are suitable for detecting leakage of 25 gpm or less into the monitored areas. Main steamline radiation will be monitored by the process radiation monitoring system (PRMS). The PRMS trip functions include main steam isolation valve isolation and reactor scram. Control room readouts and alarms will be provided by the PRMS.

In the turbine building

Reactor coolant leakage within the turbine building will be detected by monitoring main steamline pressure, main condenser vacuum, and turbine building ambient temperature in areas traversed by the main steamlines. These monitors will alarm and indicate in the control room, and, in some cases, provide signals for isolating appropriate systems.

Gross Leakage External to Drywell

Large leaks external to the drywell (or inside the drywell) will be detected by monitoring main steamline flow rate, reactor vessel level, and RCIC steamline flow rate. Abnormal conditions from any of the above monitors will alarm in the control room, and isolation of appropriate system(s) will be initiated.

Total Leakage Rate

The total leakage rate consists of all leakage, identified and unidentified, that will flow into the drywell and reactor building floor drain and equipment drain sumps. Current BWR Standard Technical Specifications (STS) limit the identified leakage rate to 25 gpm, the unidentified leakage rate to 5 gpm, and the combined leakage rate to 25 gpm. The ABWR STS will limit the RCPB identified leakage rate to 25 gpm and the unidentified leakage rate to no more than 5 gpm, and will thus meet Position C.9 of Rregulatory Guide 1.45.

Intersystem Leakage

The ABWR design provides for monitoring intersystem leakages (i.e., leakages from the reactor coolant system into other connected systems). Specifically, radiation monitors will be used to detect reactor coolant leakage into the reactor building cooling water (RBCW) system, which will provide cooling water to the RHR heat exchangers, the RIP heat exchangers, the RWCU nonregenerative heat exchangers, and the fuel pool cooling heat exchangers. At least two (up to four) process radiation monitoring channels will monitor leakage into each of the two common cooling water headers that will receive RBCW return flow from the above heat exchangers. Each channel will alarm on high-radiation conditions that indicate process leakage into the RBCW system.

Other intersystem leakage applicable to the ABWR, such as that from the RHR, HPCF, and RCIC systems, is highly unlikely, since this leakage would have to occur through closed check valves and/or closed containment isolation valves. All potential intersystem leakage from the RCPB would be into closed systems, normally filled with water. Therefore, indicators for abnormal water levels or flows in the affected areas will not be used for monitoring intersystem leakage. Thus, the ABWR design satisfies the requirements of Position C.4 of Regulatory Guide 1.29, "Seismic Design Classification," with regard to the capability for monitoring intersystem leakage.

Testing of Leak Detection Methods

The leakage detection systems are designed so as to permit operability testing and calibration during plant operation using the following methods:

- simulation of signals into trip units
- comparison of methods (e.g., airborne particulate monitoring or air cooler condensate flow versus sump fillup rate)
- comparison of channels when more than one channel is used for any one detection method (e.g., area temperature monitoring).

The above methods meet Position C.8 of Regulatory Guide 1.45.

Conclusions

The leakage detection systems proposed for the ABWR design meet Regulatory Guide 1.45, Positions C.1 through C.5, C.8, and C.9, as discussed above. They are designed to remain functional following seismic events that do not require plant shutdown. Specifically, the drywell airborne particulate radioactivity monitoring system is designed to seismic Category I standards. Failure of any system due to a safe shutdown earthquake (SSE) will not affect the particulate radioactivity monitoring system. Thus, the requirements of GDC 2, "Design Basis for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, Positions C.1 and C.2, and Regulatory Guide 1.45, Position C.6, with respect to the capability of systems to remain functional following an earthquake are satisfied.

Indicators and alarms for each leakage detection system will be provided in the control room. The level and flow from floor and equipment drain sumps, both internal and external to the drywell, will be monitored to detect leakage. Drywell air cooler condensate flows also will be monitored and will alarm in the control room on high flow. Applicants referencing the ABWR design will provide procedures and graphs to plant operators for converting the various indications (e.g., radioactivity as measured in counts per minute by the drywell airborne radiactivity monitoring subsystem) into a common leakage equivalent. Thus, the leak detection systems meet Position C.7 of Regulatory Guide 1.45.

On the basis of the above, the staff concludes that the leakage detection systems provided to detect leakage from components of the RCPB furnish reasonable assurance that structural degradation that may develop in pressure-retaining components will be detected on a timely basis. Thus, corrective action can be taken before such degradation becomes sufficiently severe to jeopardize the safety of the system, or before the leakage increases to a level beyond the capability of the makeup system to replenish the loss. The RCPB leak detection systems meet SRP Section 5.2.5. These systems conform with the guidelines of Regulatory Guide 1.29, Positions C.1 and C.2, and Regulatory Guide 1.45, Positions C.1 through C.9, and satisfy the requirements of GDC 2 and 30 and are, therefore, acceptable.

5.3.1 Reactor Vessel Materials

The materials to be used for the construction of the reactor vessel and its appurtenances have been identified by specification and conform with Section III of the ASME Code. Special requirements with regard to control of the residual elements have been identified and are considered acceptable. Compliance with the above Code provisions for material specifications satisfies the quality standards of GDC 1 and 30 and 10 CFR 50.55a.

Ordinary processes will be used for the manufacture, fabrication, welding, and nondestructive examination of the reactor vessel and its appurtenances. Nondestructive examinations will be performed in addition to Code requirements. Since the applicant has certified that the requirements of Section III of the ASME Code will be complied with, the processes and examinations to be used are considered acceptable. Compliance with these Code provisions meets the quality standards of GDC 1 and 30 and 10 CFR 50.55a.

The controls to be used during all stages of welding to prevent contamination and sensitization that could cause stress corrosion cracking in austenitic stainless steel conform with the recommendations of regulatory guides as follows:

(1) The controls to prevent contamination and excessive sensitization of austenitic stainless steel satisfy the recommendations of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel. "The staff has reviews and found acceptable the alternative approaches that will be taken by the applicant referencing the ABWR design (see Section 5.2.3 of this SER). The controls to be used provide reasonable assurance that welded components will not be contaminated or excessively sensitized before or during the welding process. These controls satisfy the quality standards c ADC 1, 4, and 30 and 10 CFR 50.55a.

(2) The staff did not review the onsite cleaning and cleanliness controls for austenitic stainless steel as discussed in Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants," because the applicant has not discussed the issue. These controls will provide reasonable assurance that austenitic stainless steel components will be properly cleaned on site thus satisfying Appendix B to 10 CFR Part 50 regarding onsite cleaning of materials and components. The applicant must include this issue in the amendment to the ABWR SSAR.

When components of austenitic stainless steels are welded, Code controls will be supplemented by conformance with the recommendations of the regulatory guides as follows. The controls imposed on delta ferrite in austenitic stainless steel welds with the recommendations of Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," because the controls to be used provide reasonable assurance that the welds will not contain microcracks. These controls also satisfy the quality standards of GDC 1 and 30 and 10 CFR 50.55a and the requirement of GDC 14 regarding fabrication to prevent rapid propagating failure of the RCPB.

When components of ferritic steels are welded, Code controls will be supplemented by conformance with the recommendations of regulatory guides as follows:

(1) The controls to be imposed on welding preheat temperatures conform with the recommendations of Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," since these controls provide reasonable assurance that components made from low-alloy steels will not crack during fabrication and minimize the potential for subsequent cracking. These controls also satisfy the quality standards of GDC 1 and 30 and 10 CFR 50.55a.

(2) The controls to be imposed during weld cladding of ferritic steel components conform with the recommendations of Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components," because the process to be used provides reasonable assurance that underclad cracking will not occur during the weld cladding process. These controls satisfy the quality standards of GDC 1 and 30 and 10 CFR 50.55a.

Integrity of the reactor vessel studs and fasteners is ensured by conformance with the recommendations of Regulatory Guide 1.65, "Materials and Inspections for Reactor Vessel Closure Studs. "Compliance with these recommendations satisfies the quality standards of GDC 1, 30, and 31 and 10 CFR 50.55a and the requirements of Appendix G to 10 CFR Part 50, as detailed in the provisions of ASME Code, Sections II and III.

The staff has reviewed the fracture toughness of the reactor vessel materials, and the reactor coolant pressure boundary materials and the materials surveillance program for the reactor vessel beltline according to the acceptance criteria and references that are set forth in SRP Section 5.3.1, Paragraphs II.5, II.6, and II.7.

GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," requires that the reactor coolant pressure boundary be designed with sufficient margin to ensure that, when stressed under operating, maintenance, and test conditions, the boundary behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized. GDC 32, "Inspection of Reactor Coolant Pressure Boundary," requires that the reactor coolant pressure boundary be designed to permit an appropriate material surveillance program for the reactor pressure vessel.

The fracture toughness requirements for the ferritic materials of the reactor coolant pressure boundary are defined in Appendices G, "Fracture Toughness Requirements," and H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50.

Compliance With 10 CFR 50.55a

The edition and addenda of the ASME Code that are applicable to the design and fabrication of the reactor vessel and RCPB components are specified in 10 CFR 50.55a. The ASME Code edition and addenda that are applicable to the ABWR are specified in SSAR Tables 1.8-21 and 3.2-3.

Compliance With Appendix G to 10 CFR Part 50

The ABWR SSAR indicates that the reactor vessel will comply with the fracture toughness requirements of Appendix G to 10 CFR Part 50. However, to confirm this conclusion, the applicant referencing the ABWR design must provide fracture toughness data based on the limiting reactor vessel materials.

Appendix G, "Protection Against Non-Ductile Failures," Section III of the ASME Code will be used, together with the fracture toughness test results required by Appendices G and H to 10 CFR Part 50, to calculate the pressure-temperature limits for the ABWR reactor vessel.

The fracture toughness tests required by the ASME Code and Appendix G to 10 CFR Part 50 provide reasonable assurance that adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the reactor coolant boundary. The use of Appendix G to Section III of the ASME Code as a guide in establishing safe operating procedures and the use of the results of the fracture toughness tests performed in accordance with the ASME Code and NRC regulations will provide adequate safety margins during operating, testing, maintenance, and anticipated transient conditions. Compliance with these Code provisions and NRC regulations constitutes an acceptable basis for satisfying the requirements of GDC 31.

Compliance With Appendix H to 10 CFR Part 50

The materials surveillance program will be used to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from exposure to neutron irradiation and the thermal environment as required by GDC 32. The ABWR surveillance program, which must be comply with Appendix H to 10 CFR Part 50 and American Society for Testing and Materials (ASTM) E-185-82, "Standard Recommended Practices for Surveillance Tests for Nuclear Reactor Vessels," requires that fracture toughness data be obtained from material specimens that are representative of the limiting base weld, and heat-affected-zone materials in the beltline region. These data will permit the determination of the conditions under which the vessel can be operated with adequate margins of safety against fracture throughout its service life.

The ABWR SSAR indicates that the reactor vessel materials surveillance program will comply with the requirements of Appendix H to 10 CFR Part 50 and ASTM E 185-82. However, to confirm this conclusion the applicant referencing the ABWR design must identify the specific materials in each surveillance capsule, the capsule lead factors, the withdrawal schedule for each surveillance capsule, the neutron fluence to be received by each capsule at the time of its withdrawal, and the vessel end-of-life peak neutron fluence.

The materials surveillance program, required by Appendix H to 10 CFR Part 50, will provide information on the effects of irradiation on material properties so that changes in the fracture toughness of the material in the ABWR reactor vessel beltline can be properly assessed and adequate safety margins against the possibility of vessel failure can be provided.

Compliance with Appendix H to 10 CFR Part 50 and ASTM E-185-82 ensures that the surveillance program will be capable of monitoring radiation-induced changes in the fracture toughness of the reactor vessel material and satisfies the requirements of GDC 32.

There should be reasonable assurance that the surveillance program will monitor the change in the properties of the material in the beltline region to the extent required for establishing pressure-temperature limits and to preserve the integrity of the vessel. The surveillance program will generate sufficient information to permit the determination of conditions under which the reactor vessel will be operated with an adequate margin against rapidly propagating fracture throughout its service lifetime.

Conclusions

The staff concludes that the reactor vessel materials are acceptable and meet the requirements of the GDC 1, 4, 14, 30, 31, and 32 of Appendix A to 10 CFR Part 50; the material testing and monitoring requirements of Appendices B, G, and H to 10 CFR Part 50; and the requirements of 10 CFR 50.55a.

5.3.2 Pressure-Temperature Limits

The acceptance criteria for reviewing the pressure-temperature limits are set forth in SRP Section 5.3.2 and 10 CFR Part 50 Appendices G and H. Appendices G and H describe the operating conditions that require pressure-temperature limits and provide the bases for these limits. These appendices specifically require that pressure-temperature limits provide safety margins at least as great as those recommended in ASME Code, Section III, Appendix G. Appendix G to 10 CFR Part 50 requires additional safety margins whenever the reactor core is critical (except for low-level physics tests) for the materials in the closure flange and beltline regions.

The staff reviewed the pressure-temperature limits that will be imposed on the reactor coolant pressure boundary during the following operations and tests to ensure that they will provide adequate safety margins against nonductile behavior or rapidly propagating failure of ferritic components as required by GDC 31:

- preservice hydrostatic tests
- inservice leak and hydrostatic tests
- heatup and cooldown operations
- core operation.

Appendices G and H to 10 CFR Part 50 require the applicant to predict the amount of increase in reference temperature, RT_{NDT}, resulting from neutron irradiation. The increase in RT_{NDT} resulting from neutron irradiation is then added to the initial RT_{NDT} and the margin to establish the adjusted reference temperature. The staff's recommended method for calculating the increase in RT_{NDT} resulting from neutron irradiation is contained in Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The recent revision of this guide (Revision 2) was published in May 1988. This revision of the guide provides a graph, tables, and formulae for calculating the increase in RT_{NDT} resulting from neutron irradiation. The relationships contained in the guide were derived by statistical analysis of 216 material data points that were reported from the testing of irradiated materials.

These materials were contained in surveillance capsules and had been irradiated inside U.S. commercial nuclear reactor vessels. As more surveillance data become available, this guide may need additional revision, since the relationship between the increase in RT_{NDT} and neutron fluence is empirically derived from analysis of material surveillance data from U.S. commercial nuclear reactor vessels.

The pressure-temperature curves in Figure 5.3-1 of the SSAR are system hydrotest limits with fuel in the vessel, non-nuclear heating limits, and nuclear (core critical) limits. These three limits represent limits that are different from SRP Section 5.3.2 guidelines. The SRP guidelines require limits on preservice hydrostatic tests, inservice leak and hydrostatic tests, heatup and cooldown operations, and core operation. The applicant needs to explain why the pressure-temperature limits in the SSAR are different from those of SRP Section 5.3.2. The pressure-temperature curves in the SSAR are generic and are not valid for any specific effective full-power years. The staff finds that the general shape of the curves is acceptable, but the curves should be used as a reference only.

For approval of any specific plant license, the applicant must submit plant-specific calculations of RT_{NDT}, stress intensity factors, and pressure-temperature curves similar to those in Regulatory Guide 1.99, Revision 2, and SRP Section 5.3.2. In response to the staff's questions, the applicant submitted a calculation of the RT_{NDT} shift of the vessel plate and weld metal. The calculation followed Regulatory Guide 1.99, Revision 2, closely. The calculated RT_{NDT} was low: for the weld metal the RT_{NDT} shift was 28°F and for the plate, 8.03°F. The reason for this low shift is the low neutron fluence and the low copper and nickel contents used in the calculation.

GE predicted the neutron fluence at end of life to be 6×10^{17} neutrons per square centimeter (n/cm^2), which is low in comparison to that of the existing BWR. The applicant will have to submit additional information during the final design approval review to show how 6×10^{17} n/cm^2 was predicted.

The staff concludes that the pressure-temperature limits imposed on the reactor coolant system for operating and testing conditions to ensure adequate safety margins against nonductile or rapidly propagating failure may be in conformance with the fracture toughness criteria of Appendix G to 10 CFR Part 50 and Section III, including Appendix G, of the ASME Code. The change in fracture toughness requirements of the pressure vessel during operation will be determined by Appendix H to 10 CFR Part 50. The use of operating limits, based on the criteria in SRP Section 5.3.2, provides reasonable assurance that nonductile or rapidly propagating failure will not occur and constitutes an acceptable basis for satisfying the requirements of 10 CFR 50.55a and GDC 1, 14, 31, and 32 of Appendix A to 10 CFR Part 50.

5.3.3 Reactor Vessel Integrity

Although most areas are reviewed separately in accordance with the staff's review plans, reactor vessel integrity is of such importance that a special summary review of all factors relating to reactor vessel integrity is warranted. The staff reviewed the fracture toughness of the ferritic materials to be used for the reactor vessel and the reactor coolant pressure boundary, the pressure-temperature limits for operation of the reactor vessel, and the materials surveillance program for the reactor vessel beltline. The acceptance criteria and references that are the bases for this evaluation are set forth in SRP Section 5.3.3, Paragraphs II.2, II.6, and II.7.

The staff has reviewed the information in each area to ensure that no inconsistencies exist that would reduce the certainty of vessel integrity. The areas reviewed and the sections of this report in which they are discussed are:

(1) reactor coolant pressure boundary materials (design) (Section 5.2.3)

(2) inservice inspection and testing of reactor coolant pressure boundary (Section 5.2.4)

(3) reactor vessel materials (fabrication methods) (Section 5.3.1)

(4) pressure-temperature limits (operating conditions) (Section 5.3.2).

The staff has reviewed all the factors contributing to the structural integrity of the reactor vessel and concludes that the reactor vessel will be materially and structurally safe. The basis for its conclusion is that the design, materials, fabrication, inspection, and quality assurance requirements for the plant will conform to applicable NRC regulations and regulatory guides and to the rules of ASME Code, Section III. The stringent fracture toughness requirements of the regulations and ASME Code, Section III, will be met, including requirements for the surveillance of vessel material properties throughout its service life, in accordance with Appendix H to 10 CFR Part 50. Also, operating limitations on temperature and pressure will be established for this plant in accordance with Appendix G to ASME Code, Section III, and Appendix G to 10 CFR Part 50.

The integrity of the reactor vessel is ensured because the vessel

(1) will be designed and fabricated to the high standards of quality required by the ASME Code and any pertinent Code cases

(2) will be made from materials of controlled and demonstrated high quality

(3) will be subjected to extensive preservice inspection and testing to provide assurance that the vessel will not fail because of material or fabrication deficiencies

(4) will be operated under conditions and procedures with protective devices that provide assurance that the reactor vessel design conditions will not be exceeded during normal reactor operation, maintenance, testing, and anticipated transients

(5) will be subjected to periodic inspection to demonstrate that the high initial quality of the reactor vessel has not deteriorated significantly under service conditions

(6) may be annealed to restore the material toughness properties if this becomes necessary

(7) will be subjected to surveillance to account for neutron irradiation damage so that the pressure-temperature limits may be adjusted

The staff concludes that the structural integrity of the reactor vessel is acceptable and meets the requirements of GDC 1, 4, 14, 30, 31, and 32 of Appendix A to 10 CFR Part 50; Appendices B, G, and H to 10 CFR Part 50; and 10 CFR 50.55a.

5.4.1 Reactor Recirculation System

The Reactor Recirculation System is not addressed in the SRP. Since the ABWR reactor recirculation system design is unique compared with the present BWR designs, the following section is provided.

A dramatic change in the ABWR design from current BWR designs is the elimination of the external loops and the incorporation of reactor internal pumps (RIPs) for reactor coolant recirculation. The containment volume is reduced by using RIPs instead of external pumps. Rupture of large-bore external pipes in the lower part of the reactor vessel is eliminated as the design-basis accident. There is no pipe larger than 2 inches in diameter below the core; thus, there is no fuel uncovery during a loss-of-coolant accident. This improves plant safety performance. The use of RIPs requires a larger diameter reactor vessel for pump impeller removal. However, this provides the benefits of reduced neutron flux at the vessel beltline and reduced reactor pressure rates during transients. Full-power

operation is possible with one RIP out of service, thus improving the plant availability. RIP maintenance is easier with less radiation exposure to plant personnel.

The reactor recirculation system consists of 10 RIPs with their impellers and diffusers internal to the reactor vessel. The RIPs themselves are mounted vertically onto and through the pump nozzles that are arranged in an equally spaced ring pattern on the bottom head of the reactor pressure vessel. The RIPs are singlestage, vertical pumps driven by variable-speed induction motors. The pump speed can be changed by varying the voltage and frequency output of the individual pump motor electrical power supply. The RIPs provide recirculation flow through the lower plenum and up through the lower grid, the reactor core, steam separators, and downcomers. The flow rate is variable over a range from minimum flow established by the pump characteristics to above the maximum flow required to obtain rated reactor power.

The RIP motors are variable-speed, four-pole, ac induction wet motor type. The operating speed of the pump motor depends on the variable voltage and variable frequency output of the adjustable speed drives. The RIP motors are cooled by water in the primary side of the reactor motor heat exchangers. Hot water in the secondary side of the heat exchanger is removed by the reactor cooling water system. There is one heat exchanger per motor. A clean purge flow is provided by the control rod drive system to inhibit reactor water from entering the motor cavity region, thereby preventing any buildup of impurities. Also, antireverse rotation devices are installed on the motor shaft to prevent possible damage of the motor as a result of reverse pump flow.

The acceptability of the flow coastdown characteristics to maintain fuel thermal margins during abnormal operational transients will be discussed in Section 15 of a supplement to this SER. The RIP inertia is significantly less than the pump inertia for operating BWRs (i.e., about a 0.7-second inertia time constant for the ABWR compared with about 3-5 seconds for operating BWRs). The ABWR design also employs motor generator (MG) sets as passthrough energy devices on 6 of the 10 RIPs. The MG sets, powered from two separate electrical distribution systems, are used to slow the coastdown of two groups of three RIPS, thus reducing the potential for departure from nucleate boiling on an all pump trip.

There is significant operating experience with the RIPs. In Europe, nearly 100 pumps are operating in BWRs, approaching 600 pump-years of experience, and very few major forced outages as a result of pump-related problems have been reported. Thus, RIP operation in BWRs is a proven technology.

On the basis of the discussions above, the staff concludes that the ABWR reactor recirculation system is acceptable.

5.4.6 Reactor Core Isolation Cooling System

The staff evaluated the reactor core isolation cooling (RCIC) system for conformance with the review guidelines and acceptance criteria of SRP Section 5.4.6. In the ABWR design, the RCIC system is a part of the emergency core cooling system. The initiation logic is diversified by adding a high drywell pressure input to the typical system initiation on reactor pressure vessel level 2.

The RCIC system is a high-pressure reactor coolant makeup system that will start independently of an ac power supply. The system will provide sufficient water to the reactor vessel to cool the core and to maintain the reactor in a standby condition if the vessel becomes isolated from the main condenser and experiences a loss of feedwater flow. The system is also designed to maintain reactor water inventory, in the event of loss of normal feedwater flow, while the vessel is depressurized to the point where the residual heat removal system can function in the shutdown cooling mode.

The RCIC system will consist of a steam-driven turbine-pump unit and associated valves and piping capable of delivering makeup water to the reactor vessel through the feedwater system. Fluid removed from the reactor vessel following a shutdown from power operation will normally be made up by the feedwater system and supplemented by inleakage from the control rod drive system. If the feedwater system is inoperable, the RCIC system will start automatically when the water level in the reactor vessel reaches the level 2 (L2) trip setpoint or will be started by the operator from the control room. The system is capable of delivering rated flow within 30 seconds of initiation. Primary water supply for the RCIC system will come from the condensate storage tank, and a secondary supply will come from the suppression pool.

The RCIC system design operating parameters, noted in SSAR Figure 5.4-9b, "RCIC Process Flow Diagram," are consistent with expected operational modes.

Essential components of the RCIC system are designated seismic Category I (in accordance with Regulatory Guide 1.29) and Quality Group B (in accordance with Regulatory Guide 1.26), as will be discussed in Section 3.2 of a supplement to this report. The proposed preoperational and initial test programs are will be discussed in Section 14 of a supplement to this report.

The RCIC system will be housed within the reactor building, which will provide protection against wind, tornados, floods, and other weather phenomena. Compliance with the requirements of GDC 2 in this regard will be discussed in Section 3.8 of a supplement to this report. In addition, the system will be protected against pipe whip inside and outside the containment as required by GDC 4, as will be discussed in Section 3.6 of a supplement to this report.

The high-pressure core flooder system (HPCF) and the RCIC systems will be located in the different rooms of the reactor building for additional protection against common-mode failures. Different energy sources will be used for pump motivation (steam turbine for RCIC pumps, electric power for HPCF pumps) and different power systems for control power. This diversity conforms to the requirements of SRP Section 5.4.6.

To protect the RCIC pump from overheating, the RCIC system contains a miniflow line that will discharge into the suppression pool when the line to the reactor vessel is isolated. When sufficient flow to the vessel is achieved, a valve in the miniflow line will close automatically, thus directing all flow to the reactor.

The RCIC system will be equipped with a discharge line fill pump (water leg pump) that will operate to maintain the pump discharge line in a filled condition up to the injection valve. Its operation will eliminate the possibility of an RCIC pump discharging into a dry pipe and minimizes waterhammer effects. A high point vent will be provided, and the system will be vented periodically, according to the Technical Specifications, to verify that the lines are filled. The RCIC

system includes a full-flow test line with water return to the suppression pool for periodic testing. The staff requires that the Technical Specifications include a system functional test at least every refueling outage, with simulated automatic actuation and verification of proper automatic valve position. The system functional test will verify that the RCIC pump will develop a minimum flow of 800 gallons per minute.

The suction piping of the RCIC system is designed for low pressure. A relief valve is, therefore, provided to protect against overpressurization of the line from the reactor coolant pressure boundary.

Suitable provisions are provided for isolation of the RCIC system from the reactor coolant system by (1) one testable valve and a closed dc-powered valve in the RCIC system discharge line and (2) two normally open motor-operated valves with appropriate closure signals to terminate the leakge of the pipe due to a break outside containment. Inservice testing of pumps and valves will be discussed in Section 3.9.6 of a supplement to this report.

The RCIC system has controls that will shut down the system if operating conditions exceed certain limits. A leak detection system is provided to detect leakage in the RCIC system.

The RCIC system is part of the emergency core cooling systems network. In SRP Section 6.3, the following GDC are identified as the acceptance criteria: 2, 17, 27, 35, 36, and 37. The RCIC system meets the requirements of these GDCs.

Level transmitters CST and H will be supported and mounted in such a way that automatic suction transfer to the suppression pool from the nonseismic tank will take place without failure during a seismic event.

The staff evaluation of TMI-2 Action Plan Items II.K.3.13, 15, 22 and 24 will be provided in a supplement to this SER.

The RCIC system meets the guidelines of Regulatory Guide 1.1, "Net Positive Suction Head for ECCS Pumps."

The RCIC pump takes suction from the CST where water temperature is low and injects into the feedwater system where water temperatures are high providing for the possibility of thermal shock at the feedwater line injection point. GE should address this potential thermal shock issue.

To the best of the staff's knowledge, steam isolation valves F063 and F064 in currently operating BWRs are not tested with a steam pipe break downstream of these valves under actual operating conditions (pressure of 1000 psig and temperature of 546°F). There is no assurance that the isolation valves will close during a break. Generic Issue GI-87, "Failure of HPCI Steam Line Without Isolation," addresses this generic concern. Testing of the valves under the actual operating conditions will be required before use.

On the basis of the review of the drawings, component descriptions, and design criteria for the RCIC system, the staff concludes, subject to resolution of the open issues discussed above, that the design of the RCIC system conforms to the Commission's regulations and to the applicable regulatory guides. It is, therefore, acceptable.

5.4.7 Residual Heat Removal System

The staff evaluated the residual heat removal (RHR) system for conformance with the review guidelines and acceptance criteria of SRP Section 5.4.7.

The RHR system consists of three independent loops, Subsystems A, B, and C. Each loop contains a motor-driven pump, heat exchanger, piping, valves, instrumentation, and controls. Each loop will be able to take suction from either the reactor pressure vessel or the suppression pool and will be capable of discharging water to either the reactor pressure vessel or back to the suppression pool via a full-flow test line. Each of the three separate shutdown cooling loops, A, B and C, will have their own heat exchangers that will be cooled by the reactor building cooling water system. RHR subsystems B and C will be used for wetwell and drywell spraying.

The RHR system will operate in five different modes:

(1) shutdown cooling
(2) suppression pool cooling
(3) wetwell and drywell spray cooling
(4) low-pressure flooder mode
(5) fuel pool cooling

For all five modes of operation, the same major hardware components (e.g., RHR pump, heat exchanger) will be used. Shutdown cooling, drywell spray cooling, suppression pool cooling and fuel pool cooling modes will be started manually. The low-pressure flooder mode, and wetwell spray cooling will be started automatically. Modes (2), (3), and (4) are reviewed in Section 6.3 of this report and also will be reviewed in Section 6.2 of a supplement; Mode (5) will be reviewed in Section 9.1.3 of a supplement to this report.

The normal operational mode of the RHR system is the shutdown-cooling mode, which will be used to remove decay heat from the reactor core to achieve and maintain a cold shutdown condition. Shutdown cooling will be started manually when the reactor pressure vessel (RPV) is depressurized to about 135 pounds per square inch gauge. The heat removed in the RHR heat exchangers will be transported to the ultimate heat sink by the RHR reactor building cooling water system.

There will be three suction lines from the RPV for shutdown cooling. This design is an improvement compared to the present single suction line in current operating BWRs, which is vulnerable to a loss of shutdown cooling by single failure of valves in the line. RHR shutdown cooling return for RHR subsystem A will be routed to the RPV through the feedwater system. RHR shutdown cooling return for RHR subsystems B and C will be routed to the RPV directly.

Isolation between the reactor and each RHR system discharge line will be provided by a check valve within the containment and a closed motor-operated isolation valve outside the containment. The suction line will contain two closed motor-operated valves, one inside and one outside the containment. These valves also

will serve a containment isolation function. Branch Technical Position RSB 5-1 (NUREG-0800) recommends that the valves provided in the suction (F012, F013) and discharge (F047, F099) of the RHR system to isolate it from the reactor coolant system have independent and diverse interlocks to protect the RHR system. Independence and diversity of pressure interlocks on motor-operated valves at the interface between the reactor coolant pressure boundary and RHR systems will be discussed in Section 7 of a supplement to this SER.

Inservice testing of pumps and valves will be discussed in Section 3.9.6 of a supplement to this report. Relief valves will be provided in each of the low-pressure lines that will interface with the reactor coolant system to protect against overpressurization from reactor coolant system leakage.

The drywell spray lines penetrating the containment will have two closed motor-operated valves, both outside containment; the lines penetrating the suppression pool will contain one closed motor-operated valve outside the containment. No pressure isolation function will be required of these valves. Containment isolation will be discussed in Section 6.2 of a supplement to this report.

The RHR pumps are motor-driven centrifugal pumps and are sized on the basis of the low-pressure flooder mode. The available net positive suction head for the RHR pumps is adequate to prevent cavitation and to ensure pump operability in accordance with Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps." The RHR system conforms to the guidelines of Regulatory Guide 1.1. Long-term operability of the RHR pumps will be discussed in Section 3.10 of a supplement to this report.

Three discharge line fill pumps are provided to maintain the RHR discharge header full of water and pressurized to avoid waterhammer upon system initiation.

For the low-pressure flooder mode, flow will be diverted to a miniflow line and discharged to the suppression pool when low flow is sensed in the injection line. The miniflow line is designed to prevent pump overheating when the valves in the injection line are closed because the reactor vessel pressure is too high to permit injection. The valves in the miniflow lines will be closed automatically when flow in the injection line is sufficient for pump cooling, thus directing all flow to the reactor. Each train of the RHR system will be tested during

normal plant operation by pumping water from the suppression pool back into the pool. The staff will require the Technical Specifications to include (1) verification of low-pressure flooder mode operability, (2) demonstration of the ability to start each pump from the control room, and (3) performance of a system functional test without requiring coolant injection into the reactor vessel. The preoperational test program will be discussed in Section 14 of a supplement to this report.

The RHR system is designed to operate with or without offsite power. Control of the RHR system will be accomplished from the control room.

Using the system process diagrams, piping and instrumentation diagrams, system safety analyses, and component performance specifications, the staff determined that the system provided for the ABWR has the capacity to bring the reactor to cold shutdown conditions in a reasonable time.

The RHR system is designed to the seismic Category I recommendations of Regulatory Guide 1.29, as will be discussed in Section 3.2 of this report. It will be housed in the reactor building for protection against the effects of flooding, tornados, hurricanes, and other natural phenomena. Compliance with GDC 2 in this regard will be discussed in Section 3.8 of this report. Conformance with Regulatory Guide 1.26 regarding quality group classifications will be discussed in Section 3.2.2 of a supplement to this report. The containment isolation requirements of GDC 55, 56, and 57 will be discussed in Section 6.2 of this report. The systems used for cooling the RHR system must meet the requirements of GDC 44, 45, and 46, as will be discussed in Section 9.2 of a supplement to this report. Those portions of the RHR system that also are part of the emergency core cooling system are designed to operate under both normal and accident conditions. The system will be protected against missiles (will be discussed in Section 3.5 of a supplement to this report) and pipe whip (will be discussed in Section 3.6 of a supplement to this report).

Typically in the current RHR system design, a cross-tie is provided between the RHR system and the service water system to flood the containment after a loss-of-coolant accident. But in the ABWR RHR system design, this cross-tie is not provided. According to GE, the containment can be flooded using the high-pressure core flooder system taking suction from the condensate storage tank (CST). The staff assumes that the CST will not be available during a severe accident that may be initiated by a seismic event. This issue will be addressed in Section 19 of a supplement to this SER.

On the basis of its review of the drawings, components descriptions, and design criteria associated with the RHR system, the staff concludes that the design of the RHR system conforms to the Commission's regulations and to the applicable regulatory guides and is, therefore, acceptable.

5.4.8 Reactor Water Cleanup System

The staff reviewed the reactor water cleanup system description and piping and instrumentation diagrams in accordance with SRP Section 5.4.8.

The reactor water cleanup system (RWCS) will continuously remove solid and dissolved impurities from the reactor water through filter demineralizers. The single loop has two parallel pumps taking common suction through a regenerative heat exchanger (RHX) and two parallel nonregenerative heat exchangers (NRHXs) from both the single bottom head drain line and the B side of the shutdown cooling suction line of the residual heat removal system. The cooled effluent from the NRHXs will pass through the RWCS pumps to the two filter demineralizers for cleanup. RWCS discharge will split to feedwater lines A and B.

The filter demineralizers will be of the pressure precoat type using filter aid and powdered mixed ion-exchange resins as a filter and ion-exchange medium. Spent resins cannot be regenerated and will be sluiced from the filter demineralizer unit to a backwash receiving tank from which they will be transferred to the radwaste system for processing and disposal. Resins will be discarded on the basis of filter-demineralizer performance, as indicated by monitoring effluent conductivity, differential pressure across the unit, and sample analysis.

Initial total capacity of resin that is finely ground and mixed will not be measured because separation into anion and cation components is not practical. To prevent resins from entering the reactor coolant system in the event of failure of a filter-demineralizer resin support, a strainer will be installed in the effluent line of each filter-demineralizer. Each strainer and filter-demineralizer vessel will have a control room alarm that will be energized by high differential pressure. If differential pressure will increase further from the alarm point, the filter demineralizer will automatically isolate.

In the event of low flow or loss of flow in the system, flow will be maintained through each filter demineralizer by its own holding pump. This will ensure that the precoat and resin materials will be held in place on the septam screens. Sample points will be provided in the common influent header and in each effluent line of the filter demineralizer units for continuous indication and recording of system conductivity. High conductivity will be annunciated in the main control room. The control room alarm setpoints of the conductivity meters at the inlet and outlet lines are 0.3 microsecond per centimeter (us/cm) and 0.1 us/cm, respectively (should be confirmed by the applicant). Reactor water conductivity of <0.3 us/cm is recommended by the Electric Power Research Institute (EPRI) in "BWR Normal Water Chemistry Guidelines - 1986 Revision," EPRI NP-4946-SR.

The influent sample point also will be used as the normal source of reactor coolant grab samples. Sample analysis also will indicate the effectiveness of the filter-demineralizer units.

The reactor chemistry limits outlined in Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors," Revision 1, Table 1, will be satisfied. These chemistry limits, as well as corrective action, will be established in the Technical Specifications.

The total capacity of the system is equivalent to 2 percent of rated feedwater flow. Each pump, NRHX, and filter-demineralizer is capable of 1 percent operation, with one RHX capable of 2 percent operation.

The suction line of the reactor coolant pressure boundary portion of the RWCS contains two motor-operated isolation valves, which will close automatically in response to signals from the leak detection and isolation system, actuation of the standby liquid control system, and high filter-demineralizer inlet temperature. The nonregenerative heat exchanger is sized to maintain the process temperature required for filter demineralization at times when return flow is partially bypassed to the main condenser or radwaste system even when the cooling system capacity of the regenerative heat exchanger is reduced.

Upon confirmation of the items indicated above, the basis for acceptance in the staff's review has been conformance of the applicant's design of the RWCS with the following:

(1) the requirements of GDC 1 by designing, in accordance with the guidelines of Regulatory Guide 1.26, the portion of the RWCS extending from the reactor vessel and recirculation loops to the outermost primary containment isolation valves to Quality Group A and by designing, in accordance with Position C.2.c of Regulatory Guide 1.26, the remainder of the RWCS outside the primary containment to Quality Group C

(2) the requirements of GDC 2 by designing in accordance with Positions C.1, C.2, C.3, and C.4 of Regulatory Guide 1.29, the portion of the RWCS system extending from the reactor vessel and recirculation loops to the outermost primary containment isolation valves to seismic Category I

(3) the requirements of GDC 14 by conforming to the positions of Regulatory Guide 1.56 in maintaining reactor water purity and material compatibility to reduce corrosion potential; thus reducing the probability of reactor coolant pressure boundary failure

(4) the requirements of GDC 60 and 61 by designing a system containing radioactivity by confining, venting, and collecting drainage from the RWCS components through closed systems.

Conclusion

On the basis of the above evaluation and confirmation of the items indicated above, the staff concludes that the RWCS conforms to the relevant requirements of GDC 1, 2, 14, 60, and 61 and the appropriate sections of Regulatory Guides 1.26, 1.29, and 1.56 (Revision 1) and, therefore, is acceptable.

6. ENGINEERED SAFETY FEATURES

6.1 Engineered Safety Features Materials

6.1.1 Metallic Materials

To meet the requirements of General Design Criteria (GDC) 1, 14, and 31 of Appendix A to 10 CFR Part 50 and 10 CFR 50.55a and to ensure the engineered safety features (ESF) materials perform the necessary safety function, the ASME Code and industry standards should be satisfied. Table 5.2-4 of the ABWR SSAR lists the pressure-retaining materials and materials specifications for the reactor coolant pressure boundary (RCPB) components. Table 6.1-1 of the SSAR lists the pressure-retaining materials and materials specifications for the primary containment system, emergency core cooling systems and their auxiliary systems, and the standby liquid control system. The ESF materials should satisfy GDC 1, 14, and 31 and 10 CFR 50.55a to ensure low probability of leakage, of rapidly propagating failure, and of gross rupture. To do so, GE should state that the ESF materials selected satisfy Appendix I to Section III of the ASME Code and Parts A, B, and C of Section II of the Code.

To meet the requirements of GDC 4, 14, and 41, the water used in the ESF systems should be controlled to provide assurance that stress corrosion cracking of unstabilized austenitic stainless steel components will not occur. In regard to the water used for emergency core cooling systems and spray systems, the applicant should follow the EPRI report, "BWR Water Chemistry Guidelines," NP-3589-SR-LD, April 1985. In light of intergranular stress corrosion cracking, the staff recommends that the applicant follow the EPRI report, "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations - 1987 Revision," NP-5283-SR-A, September 1987.

SSAR Sections 6.1.1.1.2, "Compatibility of Construction Materials With Core Cooling Water and Containment Sprays," and 6.1.1.2, "Composition, Compatibility and Stability of Containment and Core Spray Coolants," should adhere to the above recommendations.

SSAR Section 6.1.1.1.3, "Controls for Austenitic Stainless Steel," adheres to the materials specifications in SSAR Section 5.2.3.4, "Fabrication and Processing of Austenitic Stainless Steels of the Reactor Coolant Pressure Boundary Materials." The staff has reviewed and accepted Section 5.2.3.4; therefore, the staff concludes that Section 6.1.1.1.3 satisfies GDC 4 and 14 and Appendix B to 10 CFR Part 50.

The controls to be placed on concentrations of leachable impurities in nonmetallic thermal insulation used on components of the ESF comply with the guidelines of Regulatory Guide 1.36, "Nonmetallic Thermal Insulation of Austenitic Stainless Steels." The ESF thermal insulation specifications, therefore, meet the requirements of GDC 1, 14, and 31.

The staff concludes that the EFS metallic materials satisfy the requirements of GDC 1, 4, 14, 31, 35, 41, and 51 and 10 CFR Part 50.

6.1.2 Protective Coating Systems (Paints) - Organic Material

The staff conducted this evaluation to verify that the protective coatings that will be applied inside the containment conform to the testing requirements of American National Standards Institute (ANSI) Standard N101.2-1972, "Protective Coatings (Paints) for Light Water Reactor Containment Facilities," and the quality assurance guidelines of Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coating Applied to Water-Cooled Nuclear Power Plants." Compliance with these requirements provides assurance that the protective coatings will not fail under design-basic-accident conditions and generate significant quantities of solid debris that would adversely affect the engineered safety features.

The staff's review was performed in accordance with SRP Section 6.1.2. The coating system materials to be used on the exposed surfaces within the drywell will be qualified in accordance with ANSI Standard N101.2 will be used. The protective coating system for the containment will be applied in accordance with Regulatory Guide 1.54, "Quality Assurance Requirements For Protective Coatings Applied To Water-Cooled Nuclear Power Plant." The applicant has

indicated that relatively small areas within the drywell will not conform to Regulatory Guide 1.54. Any applicant referencing the ABWR will have to provide the following:

> Indicate the total amount of protective coatings and organic materials used inside the containment that do not meet the requirements of ANSI N512 (1972) and Regulatory Guide 1.54. Evaluate the generation rate as a function of time of combustible gases that can be formed from these unqualified organic materials under DBA conditions. Also evaluate the amount (volume) of solid debris that can be formed from these unqualified organic materials under DBA conditions that can reach the containment sump. Provide the technical basis and assumptions used for this evaluation.

The effects of solid debris that potentially could be formed from unqualified paints are reviewed in Section 6.2.2 of this report. The control of combustible gases that potentially could be generated from the organic materials and from qualified and unqualified paints will be reviewed in Section 6.2.5 of a supplement to this report.

The protective coating system is an open item until the designated additional information is submitted by GE and evaluated by the staff.

6.2 Containment Systems

(This section will be provided in a supplement to this SER)

6.2.1 Containment Functional Design

(This section will be provided in a supplement to this SER)

6.2.3 Containment Heat Removal System

(This section will be provided in a supplement to this SER)

6.2.4 Containment Isolation System

(This section will be provided in a supplement to this SER)

6.2.5 Combustible Gas Control

(This section will be provided in a supplement to this SER)

6.2.6 Containment Leakage Testing

The staff reviewed GE's containment leakage testing program for compliance with the containment leakage testing requirements in Appendix J to 10 CFR Part 50. Such compliance provides adequate assurance that the containment leaktight integrity can be verified throughout the service lifetime. The leakage rates will be checked periodically during service on a timely basis to ensure that such leakage will be maintained within the specified limits. Maintaining containment leakage rates within limits provides reasonable assurance that, in the event of a licensing-design-basis radioactivity release within the containment, the loss of the containment atmosphere through potential leak paths will not be in excess of the limits specified for the site.

The staff reviewed the containment leakage testing program to ensure that the containment penetrations and system isolation valve arrangements are designed to satisfy the containment integrated and local leakage rate testing requirements of Appendix J.

Type A Test

Preoperational integrated leakage rate Type A tests will be conducted in accordance with 10 Part CFR 50, Appendix J, after the primary reactor containment has been constructed and installation and local leak tests of all mechanical, fluid, electrical, and instrumentation systems penetrating the containment pressure boundary have been performed.

The objectives of the preoperational integrated leakage rate tests are to

(1) verify that the integrated leakage rate does not exceed the containment desing-basic-accident leakage rate, L_a, which is 0.5 percent by weight of the containment atmosphere in 24 hours, at peak containment accident pressure, P_a,

(2) establish a minimum allowable leakage rate, L_t, at reduced pressure, P_t, which will be used during subsequent integrated leakage rate tests

(3) obtain data that may be used to develop the leakage rate characteristics and history of the containment system

The preoperational integrated leakage rate test will be performed at both the reduced pressure, P_t, and the peak containment accident pressure, P_a. P_t will be chosen such that it will be greater than 0.5 P_a. After the preoperational integrated leakage rate test, a set of three Type A tests will be performed at approximately equal intervals during each 10-year service period. The third test of each set will coincide with the end of each 10-year major inservice inspection shutdown. The total measured leakage rate, L_{tm}, at reduced pressure P_t shall not exceed 0.75 L_t as established by the inital integrated leak rate test (ILRT).

In conducting the Type A containment integrated leakage rate test, certain systems that may operate under postaccident conditions and are normally filled with water need not be vented to the containment atmosphere. In addition, systems required to function during the Type A test shall be operable in their normal mode and need not be vented; however, the local leak rate test (LLRT) results for such systems should be added to the Type A test results. The applicant shall confirm that the LLRT results will be added. All other piping lines will be vented and drained for the Type A test.

On the basic of the above and information in the ABWR Standard SSAR regarding other acceptance criteria for the ILRT and the supplemental verification test, the staff finds the applicant's Type A test program acceptable except as noted below. The applicant has not identified (1) the systems that will not be vented or drained during the performance of the ILRT and the reasons and (2) the systems that will be vented or drained during the ILRT. Until the above information is provided and found acceptable, this issue is an open item.

Type B Test

Containment penetrations whose designs incorporate resilient seals, gaskets, or sealant compounds; piping penetrations fitted with expansion bellows serving as the containment boundary; air-lock door seals; equipment and access doors

with resilient seals; and other testable penetrations will be leak tested during preoperational testing and thereafter at periodic intervals during the lifetime of the unit in accordance with Appendix J to 10 CFR Part 50. The leak test will be performed to ensure the continuing integrity of the penetrations.

To facilitate local leak testing, GE has proposed a permanently installed system consisting of a pressurized gas source (nitrogen or air) and the manifolding and valving necessary to subdivide the testable penetrations into groups of two to five. Each group would then be pressurized, and if any leakage was detected (by pressure decay or flow meter), individual penetrations would be isolated and tested until the source and nature of the leak were determined.

The combined leakage rate of all components subject to Type B and Type C tests (described below) shall not exceed 60 percent of L_a. Type B tests shall be performed at peak containment accident pressure during each reactor shutdown for major fuel reloading, but in no case at intervals greater than 2 years. Air locks that are opened frequently during periods when containment integrity is required will be seal tested at 15 pounds per square inch gauge at selected intervals not to exceed 3 days in accordance with Appendix J.

Since the intent of the Appendix J testing program has never been to require a forced reactor shutdown just to conduct such tests within preset test intervals, the applicant shall either (1) clarify whether provisions for conducting all Type B tests at power exist in the ABWR design, or (2) request an exemption with supporting justification from the requirement for conducting Type B tests at 2-year intervals.

It also is not clear from Amendment 3 of the ABWR SSAR (1) what is basis for choosing 15 psig or the airlock seal testing pressure (2) whether the entire air lock will be Type B leak tested before initial fuel loading and at least once every 6 months thereafter at an internal pressure of not less than P_a, and (3) at what intervals the associated inflatable seals will be leak tested. Furthermore, the applicant has not provided (1) the acceptance criteria for testing the air locks and the associated inflatable seals, (2) a list of all containment penetrations that will subject to Type B tests, and (3) a list of

all penetrations that will be excluded from Type B tests, if any, and the rationale for such exclusions. On the basis of the above, the staff cannot conclude that the proposed Type B testing program for the ABWR is acceptable. Therefore, the above issues will remain as open items.

Type C Test

All primary containment isolation valves whose seats will be exposed to the containment atmosphere after a loss-of-coolant-accident will be Type C tested pneumatically with air or nitrogen at P_a. Valves that will be sealed by water will be leak tested with water as the test medium. The test pressure will be applied in the same direction as that when the valve is required to perform its safety function, unless it can be shown that results from tests with pressure applied in a different direction are equivalent or more conservative. Type C tests will be performed by local pressurization using either the pressure decay method (for pneumatic testing) or the flowmeter method (both for pneumatic testing and testing with water).

The applicant has not provided a response to the staff's request for additional information dated July 7, 1988, on a number of issues. These issues include Type C test interval; test pressure for main steam isolation valve; testing methodology for emergency core cooling system isolation valves; testing procedures for valves not covered by Appendix J procedures and a list of such valves; and lists of (1) all primary containment isolation valves that will be Type C leak tested, (2) all valves that will be hydrostatically tested and the test pressure, and (3) all valves that will be tested in the reverse direction and the justification for such testing. Further, it is not clear whether lines that contain valves that do not have 30-day water leg seals will be drained and the valves then pneumatically tested as required. Until the above issues are resolved, the staff cannot conclude that the proposed Type C test program for the ABWR is acceptable.

Conclusion

Special testing that will be performed following containment modification and inspection and reporting of test results will comply with applicable Appendix J

requirements. Besides the open items identified under Type A, B, and C tests, the applicant has not indicated whether the test, vent, and drain connections used to facilitate LRTs and LLRTs will be kept closed under administrative control during normal plant operations and will be subject to periodic surveillance to ensure their integrity and to verify the effectiveness of administrative controls. Closed loops outside the containment are relied on as containment isolation barriers for some engineered safety features (ESF) system containment penetrations. The staff will require GE to commit to include leak testing closed ESF systems outside the containment, unless it can be show that the integrity of the applicable systems will be maintained during normal plant operations. The applicant has not provided information requested by the staff on July 7, 1988, on secondary containment inleakage and potential bypass paths. The applicant has not indicated whether the bypass paths will be leak tested as specified in Branch Technical Position (BTP) CSB 6-3, "Determination of Bypass Leakage Paths in Dual Containment Plants (NUREG 0800)." The applicant has not indicated how potential contributions from the hydrogen recombiner systems will be factored into the ILRT results. Until the above issues are satisfactorily resolved, the staff cannot conclude that the proposed containment leakage testing program for the ABWR satisfies SRP Section 6.2.6.

6.3 Emergency Core Cooling System

The staff performed its evaluation of the emergency core cooling system (ECCS) in conformance with the review guidelines and acceptance criteria of SRP Section 6.3 (NUREG-0800).

The ECCS is designed to provide coolant inventory to the reactor coolant system in the event of a loss-of-coolant-accident (LOCA) in the pressure boundary. The ECCS capability extends to failures as large as a double-ended rupture of the largest piping carrying water or steam and spurious safety/relief valve operation.

6.3.1 System Description

The ECCS consists of the following:

(1) reactor core isolation cooling (RCIC) system
(2) high-pressure core flooder (HPCF) system
(3) automatic depressurization system (ADS)
(4) low pressure flooder (LPFL) system

Unlike current BWR designs, the RCIC system in the ABWR is a part of the ECCS. The initiation logic is diversified by adding a high drywell pressure input to the typical system initiation on reactor pressure vessel level 2.

The RCIC system is a high-pressure reactor coolant makeup system that will start independently of the ac power supply. The system will provide sufficient water to the reactor vessel to cool the core and to maintain the reactor in a standby condition if the vessel becomes isolated from the main condenser and experiences a loss of feedwater flow. The system is also designed to permit complete plant shutdown under conditions of loss of normal feedwater flow by maintaining reactor water inventory until the vessel is depressurized to the point where the residual heat removal RHR system can function in the shutdown cooling mode.

The RCIC system will consist of a steam-driven turbine-pump unit and associated valves and piping capable of delivering makeup water to the reactor vessel through the feedwater system. Fluid removed from the reactor vessel following a shutdown from power operation will normally be made up by the feedwater system and supplemented by inleakage from the control rod drive system. If the feedwater system is inoperable, the RCIC system will start automatically when the water level in the reactor vessel reaches the level 2 (L2) trip setpoint or will be started by the operator from the control room. The system is capable of delivering rated flow within 30 seconds of initiation. Primary water supply for the RCIC system will come from the condensate storage tank, and a secondary supply will come from the suppression pool.

The staff's evaluation of the RCIC system is given in Section 5.4.6 of this SER.

The HPCF system is provided to maintain the reactor vessel water level above the top of the active core in the event of breaks in pipes that are 1-inch in diameter or smaller and to provide cooling for large-pipe breaks in case the core is uncovered. Actuation of the HPCF system will not require the depressurization of the reactor vessel. The HPCF system is composed of two HPCF loops. Each loop includes a single motor-driven centrifugal pump that will take suction from the condensate storage tank or the primary containment suppression pool. An automatic switching feature that is based on indication of condensate storage tank and level is provided . HPCF flow will be dependent on the pressure differential that will exist between the reactor system and the suction source. The process diagram indicates that the rated HPCF flow (3200 gpm) will attained above 100 psid, which is consistent with accident analysis assumptions. The HPCF system is designed to operate from normal offsite auxiliary ac power or from diesel generators. Both HPCF pumps will be powered from two different diesel generators. The system will initiate automatically an either low water level (L1.5) or high drywell pressure signals. The system can also be placed in operation manually.

If the RCIC and HPCF systems cannot maintain the reactor water level, the ADS, which is independent of any other ECCS, will reduce the reactor pressure so that flow from the RHR system operating in the low pressure flooder mode will enter the reactor vessel in time to cool the core and limit fuel cladding temperature.

The ADS will use 8 of the 18 nuclear system pressure relief valves to relieve high pressure steam to the suppression pool. The staff's evaluation of the pressure relief valves is given in Section 5.2.2 of this SER. The ADS will be actuated when the following conditions will be satisfied: (1) drywell high pressure, (2) reactor low low water level (level-1), and (3) a permissive signal of RHR or HPCF pump discharge pressure high. A time delay of 29 seconds will be used so that the ECCS pumps will have time to restore water level, thus preventing actuation of the ADS. The instrumentation and controls for the ADS will be discussed in Section 7.3 of a supplement to this SER.

The LPFL system will to replace reactor vessel water inventory following large-pipe breaks. The system is an operating mode of the RHR system, which

consists of three independent loops (A, B, and C). Each loop has a motor-driven pump (4200 gpm), which will take suction from the suppression pool and supply water to the reactor vessel. RHR loops A, B, and C have heat exchangers that will be cooled by the RHR reactor building cooling water system and will be used to transfer the decay heat from the reactor core to the ultimate heat sink. The three LPFL (RHR) pumps will be powered from ac power buses having standby power source backup supplies. RHR pumps A, B, and C will receive emergency power from the three separate diesel generators. The RHR system valve logic will require LPFL system alignment in the event of a loss-of-coolant-accident (LOCA). The LOCA event takes precedence over other RHR system functional modes. The system will initiate automatically on either low water level (L1) or high drywell pressure signals. The reactor must be depressurized below reactor low-pressure permissive before LPFL actuation will occur.

Each of the two high-pressure core flooding loops and two of the three low-pressure flooding loops will discharge water into the core via a separate overhead flooder sparger. The A low-pressure flooding loop will discharge into the reactor pressure vessel via the feedwater system. Internal vessel piping will connect each sparger to the vessel nozzle. Compared with the current BWR core spray design, the flooder design reduces possible personnel radiation exposure because of the peripheral location of the flooder, which will minimize the need for work over the fuel during inservice inspection.

6.3.2 Evaluation of Single Failures

The staff reviewed the system description and piping/instrumentation drawings to ensure that abundant core cooling will be provided during the injection phase with and without offsite power and assuming a single failure as required by GDC 35. A low reactor vessel water level and/or high containment pressure signal will be required to start pumps and open discharge valves.

In SSAR Section 6.3.3, the applicant provided an analysis to demonstrate that the most limiting break size, break location, and single failure had been considered for the ABWR. The most limiting combinations are given in Table 6.1.

6.3.3 Qualification of Emergency Core Cooling System

The ECCS is designed to meet seismic Category I requirements in compliance with Regulatory Guide 1.29, as will be discussed in Section 3.2 of a supplement to this report. It will be housed in structures designed to withstand seismic events, tornados, floods, and other phenomena in accordance with the requirements of GDC 2, as will be discussed in Section 3 of a supplement to this report. ECCS equipment is designed in compliance with Regulatory Guide 1.26, as will be discussed in Section 3.2 of a supplement to this report.

Protecting the ECCS against pipe whip and discharging fluids in compliance with the requirements of GDC 4 and Regulatory Guide 1.46 will be discussed in Section 3.6 of a supplement to this report. Evaluation of the instrumentation and controls for the ECCS will be evaluated in Section 7.3 of a supplement to this report. Compliance with the inservice inspection requirements of GDC 36 is discussed in Section 6.6. Environmental qualification of the ECCS equipment for operation under normal and accident conditions, as required by GDC 4, will be discussed in Section 3.11 of a supplement to this report.

The available net positive suction head for the pumps in the ECCS should have adequate margin to prevent cavitation and ensure pump operability in accordance with Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps." The ECCS system pumps should meet Regulatory Guide 1.1.

TABLE 6.1

SINGLE FAILURE EVALUATION

Assumed failure	Systems Remaining			
	Automatic Depressurization (ADS)	Reactor Core Isolation Cooling (RCIC)	High-Pressure Core Flooder (HPCF)	Residual Heat Removal Low-Pressure Flooder (LPCF)
Emergency diesel (generator A)	all	1	2	2
Emergency diesel (generator B or C)	all	1	1	2
RCIC injection valve	all	-	2	3
One ADS valve	minus one	1	2	3

The LPFL (RHR) system is not designed to withstand normal reactor operating pressure. Each of the low-pressure lines that will discharge into the reactor coolant system has a testable check valve inside the primary containment backed up by a normally closed motor-operated gate valve outside the containment. Relief valves are provided in the low-pressure lines to protect against leakage from the reactor coolant system. An interlock that prevents their opening until the reactor coolent pressure is below the low-pressure ECCS design pressure is provided on the motor-operated valves.

Containment isolation in accordance with the requirements of GDC 55 will be discussed in Section 6.2 of a supplement to this report. The periodic testing and leak-rate criteria for those valves that isolate the reactor system from the ECCS will be discussed in Section 3.9.6 of a supplement to this report. The detection of leaks from those portions of the ECCS within the primary coolant pressure boundary is discussed in Section 5.2.5 of this report.

To protect the pumps from overheating, all the ECCS pumps have miniflow lines to permit a limited amount of flow if an isolation valve between the reactor coolant system and the ECCS is closed for any reason. When flow in the injection lines is sufficient for pump cooling, valves in the miniflow lines will shut automatically, diverting all flow to the pressure vessel. Each LPFL pump suction line from the suppression pool has an open motor-operated valve outside the containment. The suction of the HPCF from the suppression pool contains a closed motor-operated valve so the HPCF will initially draw water from the condensate storage tank. When the condensate storage tank water is exhausted, the suppression pool suction valve will be opened automatically. Isolation of the suppression pool from the reactor building in accordance with GDC 56 will be discussed in Section 6.2 of a supplement to this report.

As a backup to the high-pressure core flooder system, the automatic depressurization system can be used to depressurize the system and allow the low-pressure cooling systems to function if there should be a small break. The nitrogen supply to the valves of the automatic depressurization system will be provided by seismically qualified accumulators.

One of the design requirements of the emergency core cooling system is that cooling-water flow be provided rapidly following the initiation signal. By always keeping the ECCS pump discharge lines full, the lag time between the signal for pump start and the initiation of flow into the reactor pressure vessel can be minimized. In addition, full discharge lines will prevent potentially damaging waterhammer occurrences on system startup. In the ABWR a fill system consisting of a jockey pump (discharge line fill pump) is provided in the RCIC system and the RHR systems. Maintenance of the filled status of the system will be ensured by continuous indication of pump operation and pump discharge pressure. The staff requires that the uppermost vent lines in the filled systems be opened and checked for flow to eliminate the possibility of the formation of air pockets. The discharge line fill function for the HPCF system is provided statically by connection to the condensate storage tank.

The ECCS pumps must have the capability to operate for an extended period during the long-term recirculation phase following a loss-of-coolant accident. Pump operability will be discussed in Section 3.10 of a supplement to this report.

Safety/relief valve (SRV) operability will be demonstrated during the power ascension phase of the plant startup test program by manually actuating each SRV (including the ADS valves) one at a time to measure discharge capacity and to demonstrate that no blockage exists in the valve discharge line. After commercial turnover, all of the SRVs will be tested in accordance with Section XI, Article IWV, of the ASME Code. Linear variable differential transformers and thermocouples will be installed in the discharge line of each SRV to monitor valve position and SRV leakage in accordance with NUREG-0737, Item II.D.3, and will be discussed in Section 7 of a supplement to this report. Environmental qualification of the SRVs will be discussed in Section 3.11 of a supplement to this report.

6.3.4 Testing

The applicant states that ECCS operability will be demonstrated by preoperational and periodic testing, as required by Regulatory Guide 1.68 and GDC 37.

6.3.4.1 Preoperational Tests

Preoperational tests will ensure the proper functioning of controls, instrumentation, pumps, piping, and valves. Pressure differentials and flow rates will be measured for later use in determining acceptable performance in periodic tests. The applicant has committed to conform to the guidelines of Regulatory Guide 1.68 mentioned above for preoperational and initial startup testing of the ECCS, as will be noted in Section 14 of a supplement to this report.

6.3.4.2 Periodic Component Tests

The staff will require the applicant referencing the ABWR design to test the ECCS subsystems (except for the ADS) periodically to show that specified flow rates are attained. Also, the staff will require that a test be performed every refueling in which all subsystems are actuated through the emergency operating sequence.

6.3.5 Performance Evaluation

The information currently in the SSAR, the results of the break analyses, along with the results of the review of each ECCS system, are used to verify that the proposed ECCS meets the performance criteria in 10 CFR 50.46. Compliance with the first two criteria is demonstrated analytically. Compliance with the coolable geometry and long-term cooling criteria is demonstrated using engineering judgment. Coolable geometry is maintained if the first two criteria are met. Long-term cooling capability is verified by the composite review of the ECCS and the various support systems. The five major acceptance criteria for the ECCS, as specified in 10 CFR 50.46, are:

(1) The calculated maximum peak cladding temperature (PCT) shall not exceed $2200^{o}F$.

(2) The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

(3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the plenum volume were to react.

(4) Calculated changes in core geometry shall be such that the core remains amenable to cooling.

(5) The calculated core temperature shall be maintained at an acceptably low value, and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core after any calculated successful initial operation of the ECCS.

The applicant has demonstrated compliance with the first three of these criteria in Table 6.2

A coolable geometry is demonstrated by compliance with the criteria for the PCT and the maximum cladding oxidation as discussed in GE Topical Report NEDO-20566.

The staff reviewed the LOCA analyses in SSAR Section 6.3.3. The calculations were done in accordance with the methods described in GE Topical Report NEDO-24011-P A-US, Amendment 20, 'General Electric Standard Application for Reactor Fuel - United States Supplement."

The applicant used SAFER/GESTR-LOCA application methodology for the analysis of postulated loss-of-coolant accidents for the ABWR. The staff accepted this methodology generically for referencing in a letter dated June 1, 1984 from C. O. Thomas (NRC) to J. F. Quirk (GE) entitled, "Acceptance for Referencing of Licensing Topical Report NEDE-23785, Revision 1, Volume III (P), the GESTR-LOCA and SAFER Models for Evaluation of the Loss of Coolant Accident." The GE analyses included break sizes ranging from a bottom head drain line break (0.0218 square foot) to the main steamline break outside containment (4.24 square feet). Different break sizes were analyzed in conjunction with ECCS failure combinations. The cases were evaluated to establish the trend of PCT curves (Appendix K to 10 CFR Part 50 and bounding values) versus break size.

Eight break sizes are summarized in Table 6.3-4 of the SSAR. The most limiting break is the main steamline break outside the containment which results in 1149°F. This is well below the acceptance criterion of 2200°F.

The staff confirmed that the LOCA analysis methodology is consistent with and bounded by the staff's generically approved use of the methodology. The staff concluded that the PCT values, peak local oxidation values, and core-wide metal-water reaction values are well below staff acceptance values. Thus, the analyses and results are in accordance with NRC requirements, and conformance with the ECCS acceptance criteria of 10 CFR 50.46 and Appendix K has been demonstrated.

Long-term cooling is ensured by the use of redundant systems that have adequate water sources available to remove the decay heat generated within the reactor core and transfer the heat to the ultimate heat sink. No single failure was identified that would prevent the ECCS from meeting this criterion. The systems are designed to prevent any core uncovery.

The LPFL flow will be diverted automatically to wetwell spray cooling or to suppression pool cooling. If both a LOCA signal and a pool-cooling initiation signal are present, the RHR system will function in the low-pressure flooding mode. The ABWR emergency procedures should contain adequate cautions to deter the operator from premature flow diversion. These procedures, which will be based on guidelines accepted by the staff (NUREG-0737, Item I.C.1, in Section 13 of a supplement to this report), will caution the operator against diversion unless adequate core cooling is ensured. LPFL diversion will be identified in the procedure as secondary to core cooling requirements, except in those instances outside the design envelope involving multiple failures for which maintenance of containment integrity is required to minimize risk to the environment.

6.3.6 Conclusions

The staff has reviewed the piping and instrumentation drawings and the description of the ECCS in the SSAR. It finds the design of the system acceptable because it conforms to the pertinent Regulatory Guides, SRP sections, and GDC.

In addition, on the basis of the discussion above, the staff finds the performance of the ECCS meets the performance criteria in the September 16, 1988, revision of 10 CFR 50.46, Appendix K, and hence it is acceptable.

TABLE 6.2 Demonstration of Compliance with Emergency Core Cooling System Criteria

Criterion	Maximum from break analyses	Allowable
Peak cladding temperature (PCT), °F	1149	2200
Maximum cladding oxidation, %	0.03	17
Total hydrogen generation, %	0.03	1

6.4 Control Room Habitability Systems

The staff reviewed the control room habitability systems in accordance with SRP Section 6.4. Conformance with the acceptance criteria formed the basis for the staff's evaluation of the control room habitability systems with respect to the applicable regulations of 10 CFR Part 50. Specifically, the SRP acceptance criteria require the design to meet GDC 4, "Environmental and Dynamic Effects Design Bases," as it relates to accommodating the effects of and being compatible with postulated accidents, including the effects of the release of toxic gases; GDC 19, "Control Room", as it relates to maintaining the control room in a safe, habitable condition under accident conditions by providing adequate protection against radiation and toxic gases; and TMI action plan Item III.D.3.4 (NUREG-0737) requirements as they relate to providing protection against the effects of release of toxic substances, either on or off the site. Since the ABWR design is applicable only for a single unit, GDC 5, "Sharing of Structures, Systems and Components," is not applicable to the design.

The control room habitability systems will provide (1) missile protection, (2) radiation shielding, (3) radiation monitoring, (4) air filtration and ventilation, (5) lighting, (6) personnel and administrative support, and (7) fire protection. The control building heating, ventilation, and air conditioning (HVAC) system and components will be located in a seismic Category I structure that will be protected from tornado, missile, pressure and flood damage. The HVAC ducting will be ESF grade. HVAC hangers are designed to seismic Category I standards. The HVAC system will maintain the control room atmosphere temperature at a habitable level to permit prolonged personnel occupancy throughout postulated design-basis-accidents. The system design provides for control room pressurization and filtered intake during accident situations and for purging of smoke and toxic gases. The system is capable of automatic transfer from its normal operating mode to its emergency or isolation modes on detection of adverse conditions (e.g., high radiation, smoke). The system has sufficient redundancy to ensure operation under emergency conditions, assuming the single failure of any one active component. Backup power sources are provided for the essential components of the HVAC system Section 9.4.1 of a supplement to this SER will provide more information on the control room room HVAC system. The habitability systems provide the capability to detect and limit the introduction of radioactive material and smoke into the control room. The ABWR design

relies on noncombustible construction and heat - and flame-resistant materials throughout the plant to minimize the likelihood of fire and consequent fouling of the atmosphere with smoke or noxious vapor. Further, a sufficient number of individual respirators (subject to periodic operational testing) are provided to protect against instrusion of toxic gases into the control room. Nonseismic pipe, ductwork for kitchen and sanitary facilities, and other nonessential components in the control building are designed to ensure that their failure during a safe shutdown earthquake will not adversely affect essential components. Potential sources of danger such as pressure vessels and carbon dioxide fire-fighting containers will be located outside the control room and the compartments containing control building life support systems. There will be high-energy lines near the control room; therefore, the habitability systems will be protected against dynamic effects that may result from possible failures of such lines. On the basis of the above, the staff concludes that the control room habitability systems satisfy GDC 4.

In accordance with TMI Action Plan Item III.D.3.4, applicants referencing this standard design will be required to demonstrate that control room operators are adequately protected against the effects of the release of toxic substances, either on or off the site, and that the plant can be safely operated or shut down under conditions created by any design-basis accident. The amounts and locations of any possible sources of toxic substances in each plant vicinity will have to be identified following the methods outlined in Regulatory Guides 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," and 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release." Specific detectors to permit automatic control room isolation will have to be provided, where necessary. The need for site-specific toxic gas protection will be reviewed to ensure that control room operators are protected against releases of hazardous material.

Regarding the design features provided for compliance with GDC 19, the staff requested additional information on a number of issues by letter dated July 7, 1988 (e.g., makeup air inlets for the control room emergency zone, location of control room ventilation inlets relative to major potential plant release points, radiation protection, instrumentation, minimum positive pressure

during pressurization mode). Until the staff evaluates and finds the applicant's response to the above request acceptable, it cannot conclude that the control room habitability systems meets GDC 19. Therefore, demonstration of compliance with GDC 19 is an open item.

6.5.1 Engineered Safety Features Atmosphere Cleanup Systems

The staff reviewed the engineered safety features (ESF) atmosphere cleanup systems in accordance with SRP Section 6.5.1. It performed an audit review of each of the areas in the "Areas of Review" portion of the SRP section according to the guidelines in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for the staff's evaluation of the ESF atmosphere cleanup systems with respect to the applicable regulations of 10 CFR Part 50.

The ABWR design has two ESF filter systems: the control building outdoor air cleanup system and the standby gas treatment system.

Control Building Outdoor Air Cleanup (CBOAC) System

The function of the CBOAC system is to supply filtered outside air to the control building after a design-basis accident (DBA) and to pressurize the control room. This system will permit operating personnel to remain in the control room following a DBA. The applicant has committed to provide information on the system design in SSAR Section 9.4.1 later. Therefore, at this time, the staff cannot conclude that the design meets the acceptance criteria of SRP Section 6.5.1. The system design remains an open item.

Standby Gas Treatment System (SGTS)

The primary function of the SGTS is to filter and thereby reduce offsite airborne releases of radioiodine and particulates following a DBA (e.g., LOCA, fuel-handling accident, fuel cask drop accident). This will be accomplished by automatic isolation of the secondary containment from its normal HVAC air paths and automatic actuation of the SGTS on receipt of a LOCA signal or upon detec-

tion of high radiation levels in the secondary containment normal HVAC exhaust or in the refueling floor exhaust. The system can also be manually initiated from the control room to reduce airborne radioactive iodine and particulate releases when the primary containment is (1) vented during startup for pressure control, (2) purged or deinerted during normal plant operation, or (3) purged before personnel access to the primary containment during plant shutdown for refueling. The system will maintain the secondary containment at a slight negative pressure (i.e., 0.25 inch water gauge) with respect to the environs and process the effluent gases through a filter train to remove airborne iodines and particulates. The system consists of two identical, parallel, physically separated, 100-percent capacity subsystems with associated piping and ducts, valves, dampers and controls, and a common single filter train. Each subsystem has a dryer train consisting of a demister for a flow of 1200 cubic feet per minute, an electric process heater and an exhaust fan downstream of the heater designed for an air flow of 1200 cubic feet per minute. The single filter train downstream of the exhaust fans is designed for an air flow capacity of 1200 cubic feet per minute. The filter train consists of a prefilter (one bank of two filters), a high-efficiency particulate air (HEPA) filter, a 6-inch-deep charcoal adsorber, a downstream HEPA filter, and space heaters. The system is designed to seismic Category I requirements and will be housed in a seismic Category I structure.

The applicant has not provided specific information regarding (1) the system's compliance with each of the regulatory positions in Regulatory Guide 1.52, Revision 2, "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," and with each of the instrumentation items in Table 6.5.1-1 of SRP Section 6.5.1 and (2) which area process monitors (other than the fuel area ventilation exhaust monitors) will actuate the SGTS on detection of high airborne radioactivity level (e.g., emergency core cooling system/reactor water cleanup/reactor core isolation cooling equipment rooms, shield wall annulus, primary containment purge). Further, the staff is concerned about the provision of a single filter train (SRP Section 6.5.1 requires redundant filter trains) and the nonconservative sizing of the charcoal adsorber based on an assumed decontamination factor of 100 for elemental and particulate forms of radioiodine in the suppression pool

(Regulatory Guide, 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant-Accident for Boiler Water Reactor," assumes a decontamination factor of 1 for all forms of iodine, and Regulatory Guide 1.52 requires compliance with the above guide for the design of the adsorber section). Also, the applicant has not provided any analysis to demonstrate that the use of the system during the inerting, deinerting, pressure control, or purging of the primary containment during normal plant operation will not impair its functional capability during a DBA. Until the above concerns are satisfactorily resolved, the staff cannot conclude that the SGTS design meets the acceptance criteria in SRP Section 6.5.1 or implements an adequate design on an alternative defined basis. The system design remains an open item.

6.5.3 Fission Product Control Systems and Structures

The staff reviewed fission product control systems and structures in accordance with SRP Section 6.5.3. The applicant has provided two fission product control systems and structures. These are the primary containment, which includes the suppression pool with scrubbing and retention capability and the secondary containment which includes the standby gas treatment system with filtration capability.

Primary Containment

The primary containment is a cylindrical steel-lined reinforced concrete structure that will form a limited leakage boundary for fission products released to the containment atmosphere following a LOCA or any other accident releasing lesser amounts of fission products. The structure will be divided by a reinforced concrete diaphragm floor and the reactor vessel pedestal into upper and lower drywells and a suppression chamber (wetwell). The diaphragm floor will be rigidly attached to the reactor pedestal and the containment wall. The diaphragm floor includes a liner that will prevent steam bypass from the upper drywell to the suppression chamber air space during any accident. The primary containment will be totally enclosed within the reactor building, a portion of which will form the secondary containment. The applicant has assumed a design leak rate of 0.5 percent per day of the free containment volume at design

pressure. A test program will be implemented to confirm leak integrity of the primary containment structure. The primary containment will provide a passive barrier to limit leakage of airborne radioactive material following a LOCA by immediate closure of containment isolation valves except for those necessary for ECCS and ESF functions. Further information on the primary containment design, its isolation methods, isolation times, and leak tests is provided in 6.2.6 of this SER and will be provided in Sections 6.2.1.1 and 6.2.4 of a supplement to this SER. The applicant has assumed a decontamination factor (DF) of 100 for the particulate form of iodine due to scrubbing and retention in the suppression pool.

In the LOCA dose calculations, the applicant has assumed percentages of 0.03 and 99.97 with suppression pool DFs of 1 and 100 for the organic and particulate forms of the iodine source term, respectively. These assumptions are inconsistent with the assumptions in Regulatory Guide 1.3 for evaluating LOCA doses. Further, it is the staff's current position that a suppression pool DF of more than 10 for the elemental and particulate forms of iodine for Mark II and Mark III containment designs (applicable for the ABWR) will require detailed justification. Also, the applicant has assumed a primary containment leak rate of 0.25 percent per day of free containment volume 24 hours after a LOCA occurs; however, the applicant has not provided justification for the above assumption. Since the applicant has used the above nonconservative assumptions for the fission product control systems and structures in its LOCA dose evaluations, the staff cannot conclude that the systems provided in the ABWR design have the capability to reduce LOCA doses to within the 10 CFR Part 100 limits. Therefore, the above issues will remain as open items.

Secondary Containment

The secondary containment is a reinforced concrete building that will form an envelope surrounding the primary containment above the basemat. It will enclose all penetrations through the primary containment and all those systems external to the primary containment that could become a potential source of radioactive release after an accident. Following an accident, the secondary containment normal HVAC paths will be secured, and the SGTS will actuate and maintain the secondary containment at a slight negative pressure (-0.25-inch water gauge) with respect to the environs. The SGTS will filter airborne fission

products leakage (iodines and particulates) from the primary containment before their release. The applicant has assumed an inleakage rate of 50 percent of the secondary containment free volume per day at a differential pressure of -0.25-inch water gauge with respect to the environs. The secondary containment design and that SGTS will be discussed in Sections 6.2.3 and 6.5.1 of a supplement to this SER.

Besides the open items identified in Section 6.5.1 of this SER, the applicant has not provided information on the drawdown time for achieving the secondary containment design negative pressure and its effect on LOCA dose evaluations. The staff is unable to conclude that the applicant has demonstrated the 99-percent removal efficiency assumed for the SGTS filter train for all forms of iodine. The staff cannot conclude that the fission product control system provided in the form of the SGTS for the ABWR design has the capability to reduce the DBA doses to within 10 CFR Part 100 limits. Therefore, the above issues are open items.

6.6 Inservice Inspection of Class 2 and 3 Components

The review of the ABWR SSAR according to SRP Section 6.6, Paragraph II.1, has not been completed because the applicant has not submitted a complete preservice inspection (PSI) program and inservice inspection (ISI) program.

The review according to SRP Section 6.6, Paragraph II.3, will be done when the completed PSI program has been received.

The review according to SRP Section 6.6, Paragraph II.4, has not been done because the inspection interval of the PSI program was not addressed in ABWR SSAR.

The review according to SRP Section 6.6, Paragraph II.5, has been done. The applicant has incorporated ASME Code Article IWC-3000, "Acceptance Standards for Flaw Indications," into the ISI program. However, ongoing NRC generic activities and research projects indicate that the currently specified ASME Code procedures may not always be capable of detecting the acceptable-size flaws specified in these standards. For example, ASME Code procedures specified

for volumetric examinations of vessels, bolts and studs, and piping have not proved to be capable of detecting the acceptable-size flaws in all cases. The staff will continue to evaluate the development of improved procedures and will require that these improved procedures be made a part of the inservice examination requirements. The applicant's repair procedures based on ASME Code Articles IWC-4000 and IWD-4000, "Repair Procedures," have not been reviewed. Repairs are not generally necessary in the PSI program. This subject will be addressed during the review of the ISI program.

The review according to SRP Section 6.6, Paragraph II.7, has been completed. The applicant committed to examine all circumferential welds 100 percent volumetrically during each inspection interval. Augmented inservice inspection will not be required for the ABWR design because there are no guard pipes enclosing high-energy piping between containment isolation valves.

The review according to SRP Section 6.6, Paragraph II.8, has not been completed because the applicant has not provided the information specified in the SRP. Article IWC-1220 of the Code establishes criteria for exempting certain categories of welds from volumetric and surface examination. The applicant has not identified any such exemptions based on IWC-1220 in the ABWR SSAR.

The review according to SRP Section 6.6, Paragraph II.9, has not been completed because the applicant has not identified the limitations to the examination. Specific areas where ASME Code examination requirements cannot be met will be identified as the PSI progresses. The complete evaluation of the PSI program will be presented in a supplement to this SER after GE submits the required examination information and identifies all plant-specific areas where ASME Code Section XI requirements cannot be met and provides a supporting technical justification.

6.6.2 Examination Requirements

GDC 36, 39, 42, and 45 require, in part, that the Class 2 and 3 components be designed to permit appropriate periodic inspection of important components to ensure system integrity and capability. 10 CFR 50.55a(g) defines the detailed requirements for the PSI and ISI programs for light-water-cooled nuclear power

facility components. On the basis of construction permits issued on or after July 1, 1974, this section of the regulations requires that a preservice inspection program be developed and implemented using the edition and addenda of Section XI of the ASME Code applied to the construction of the particular components. However, components (including supports) may meet requirements in subsequent editions of the Code and addenda that are incorporated by reference in 10 CFR 50.55a(b), subject to the limitations and modifications listed therein. The initial ISI program must comply with the requirements of the latest edition and addenda of Section XI of the ASME Code, subject to the limitations and modifications in 10 CFR 50.55a(b).

6.6.3 Evaluation of Compliance With 10 CFR 50.55a(g)

The review of compliance with 10 CFR 50.55a(g) has not been completed because the applicant has not discussed the applicable provisions of 10 CFR 50.55a(g) in ABWR SSAR Section 6.6.

6.6.4.4 Issues To Be Addressed During Final Design Approval Review

The staff will complete its review after GE:

(1) Identifies the specific areas where the applicable ASME Code requirements cannot be met after the initial examinations are performed and provides a supporting technical justification for relief.

(2) Dockets complete and acceptable PSI and ISI program plans. The programs should include references to the edition and addenda of ASME Code, Section XI, that will be used for the selection of components for examination, lists of the components subject to examination, description of the components exempt from examination by applicable Code, and the examination isometric drawings.

(3) Plans for preservice examination of the reactor pressure vessel welds to address the degree of compliance with Regulatory Guide 1.150, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations."

(4) Discusses the near-surface examination and resolution with regard to detecting service-induced flaws and the use of electronic gating as related to the volume of material near the surface that is not being examined.

The staff considers the review of the PSI program an open issue subject to GE providing the information noted above. The staff's evaluation of the response will be reported in a supplement to this SER.

The complete ISI program has not been submitted by GE. The staff will evaluate this program based on the applicable ASME Code edition and addenda determined on the basis of 10 CFR 50.55a(b).

6.6.4 Conclusions

Compliance with the preservice and inservice inspection requirements the ASME Code and 10 CFR Part 50 constitutes an acceptable basis for satisfying the applicable requirements of GDC 36, 39, 42, and 45.

6.7 Main Steam Isolation Valve Leakage Control System

(This section will be provided in a supplement to this SER).

7 INSTRUMENTATION AND CONTROLS

(This section will be provided in a supplement to this SER.)

8 ELECTRIC POWER SYSTEMS

(This section will be provided in a supplement to this SER.)

9 AUXILIARY SYSTEMS

(This section will be provided in a supplement to this SER.)

10 STEAM AND POWER CONVERSION SYSTEM

(This section will be provided in a supplement to this SER.)

11 RADIOACTIVE WASTE MANAGEMENT

(This section will be provided in a supplement to this SER.)

12 RADIATION PROTECTION

(This section will be provided in a supplement to the SER.)

13 CONDUCT OF OPERATIONS

(This section will be provided in a supplement to this SER.)

14 INITIAL TEST PROGRAM

(This section will be provided in a supplement to this SER.)

15 TRANSIENT AND ACCIDENT ANALYSIS

(This section will be provided in a supplement to this SER.)

16 TECHNICAL SPECIFICATIONS

(This section will be provided in a supplement to this SER.)

17 QUALITY ASSURANCE

17.1 Quality Assurance During the Design Phase

17.1.1 General

The quality assurance (QA) program for the design phase of the ABWR is described in Chapter 17 of the ABWR SSAR. Chapter 17 references the General Electric Company (GE) QA topical report, "Nuclear Energy Business Operations Quality Assurance Program Description," May 1987, NEDO-11209-04A, Revision 7, which the staff has reviewed and found acceptable. Chapter 17 also provides additional QA information specifically applicable to the ABWR. The SSAR was submitted by GE, which is responsible for designing the ABWR. The staff assessed GE's QA program description for the design phase of the ABWR to determine if it complies with the requirements of 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," and with applicable QA-related regulatory guides listed in Table 7.1. The basis of the staff's review was SRP Section 17.1 (NUREG-0800).

17.1.2 Organization

The structure of the organization responsible for the design of the ABWR and for the establishment and execution of the design-phase QA program is shown in Figure 17.1. The line organizations have been assigned specific QA responsibilities, including both internal audits and audits of suppliers to ensure compliance with the QA program. These audits are superimposed by audits conducted by GE's Nuclear Quality Assurance (NQA) organization.

The General Manager of Nuclear Operations is responsible for ensuring that (1) the intent of GE's nuclear quality policy is reflected in its nuclear products and services, (2) a system is in place to independently assess the performance of organizations that affect the quality of these items, and (3) a system is in place to resolve issues that could affect GE's ability to satisfy its nuclear quality policy and other quality-related commitments.

NQA is a staff organization responsible for establishing the nuclear quality policy and procedures that are issued by the Vice President and General Manager of GE Nuclear Energy. NQA is also responsible for (1) auditing the various line organizations involved in the nuclear business and ensuring conformance of these organizations' procedures and practices with applicable corporate and nuclear quality-related policy and procedures, (2) ensuring integration of the organizations' quality planning into an effective QA program, (3) participating in management review boards that operate independently of the design verification by the line organizations, and (4) specifying how the line organizations are to comply with the nuclear quality policy and procedures. For the ABWR design, NQA is responsible for coordinating and integrating the QA program as it relates to engineering and management of the project.

A quality council aids NQA in fulfilling its responsibilities. The council's responsibility is to ensure total quality system coverage, uniformity, consistency, and continuity and to eliminate system deficiencies. The quality council is chaired by the Manager, NQA. Its members, as shown in Figure 17.1, are the managers responsible for QA in each of the major nuclear organizations. The council provides these managers direct access to top-level management and provides a forum for the review of quality problems and corrective actions.

The line organizations are responsible for planning and implementing the QA functions performed within their areas of responsibility so that each organization's QA program complies with the nuclear QA policy and procedures established by NQA. The individual QA managers report to their department-level management and have the organizational independence and authority to identify quality-related problems; initiate, recommend, or provide solutions pertaining to conditions adverse to quality; and verify implementation of such solutions.

The design of the ABWR is by GE and its major technical associates, Hitachi and Toshiba. The lead responsibility to produce each specification (through the major purchasing specifications) and drawing is assigned to one design organization within GE Nuclear Energy, Hitachi, or Toshiba. The content of each of these documents is reviewed and approved by GE engineering personnel, and GE is responsible for the design and the supporting calculations and records for the ABWR.

GE Nuclear Energy engineering organizations are responsible for the ABWR design and design control by

(1) ensuring incorporation of applicable regulatory requirements, codes, standards, criteria, and design bases into the design

(2) ensuring incorporation of project design requirements into the design

(3) translating the design information onto the appropriate design documents

(4) verifying the design adequacy either through independent design review, the use of alternative or simplified calculational methods, or the performance of a suitable testing program

(5) coordinating design activities among interfacing design engineers and design organizations

(6) reviewing, approving, issuing, and distributing design documents under a controlled document system

(7) controlling design changes and changes to design documents in accordance with documented procedures

(8) providing for the retention, storage, control, and retrievability of design record documents

(9) taking corrective action as necessary to correct design errors and to improve the design control function

17.1.3 Quality Assurance Program

GE has structured its nuclear QA program to satisfy Appendix B to 10 CFR Part 50 and the provisions of the NRC guidance shown in Table 17.1. This QA program is used for the design of the ABWR. The program is implemented by means of written policies, procedures, and instructions. These documents control quality-related activities in accordance with the requirements of Appendix B to 10 CFR Part 50 and with applicable regulations, codes, and standards. The

GE Nuclear Energy QA organizations are responsible for ensuring that procedures and instructions provide for meeting the QA requirements. In addition, QA personnel conduct reviews and audits to verify the effective implementation of the program.

GE's nuclear QA program requires that implementing documents encompass detailed controls for (1) translating codes, standards, regulatory requirements, technical specifications, engineering requirements, and process requirements into drawings, specifications, procedures, and instructions; (2) developing, reviewing, and approving procurement documents and changes thereto; (3) prescribing all quality-related activities by documented instructions, procedures, drawings, and specifications; (4) issuing and distributing approved documents; (5) purchasing items and services; (6) identifying materials, parts, and components; (7) performing special processes; (8) inspecting and/or testing materials, equipment, processes, and services; (9) calibrating and maintaining measuring and test equipment; (10) handling, storing, and shipping items; (11) identifying the inspection, test, and operating status of items; (12) identifying and dispositioning nonconforming items; (13) correcting conditions adverse to quality; (14) preparing and maintaining QA records; and (15) auditing activities that affect quality.

Training and experience requirements are defined for each position in the GE Nuclear Energy organization. In addition, GE provides for the indoctrination and training of personnel performing activities affecting quality to ensure that appropriate proficiency is achieved and maintained. The indoctrination and training are carried out through documented procedures, on-the-job training, personal contacts, and meetings. The training also ensures that personnel responsible for quality-related activities are instructed as to the purpose, scope, and implementation of the quality-related manuals, instructions, and procedures.

The ABWR design and changes to it are formally verified. Design verification is a process for an independent review of designs against design requirements to confirm that the designer's methods and conclusions are consistent with requirements and that the resulting design is adequate for its specified purpose. Design verification is performed and documented by persons other than those responsible for the design, using the method specified by the design

organization. Designs are verified by one or more of the following methods: design review, qualification testing, alternative or simplified calculations, or checking. Team design reviews are ongoing reviews of design, selected by engineering management, to evaluate design adequacy including concepts, the design process, methods, analytical models, criteria, materials, applications, or development programs. When appropriate, team design reviews are used to verify that product designs meet functional, contractual, safety, regulatory, industrial codes and standards, and GE Nuclear Energy requirements. The selection of the design review team depends on the product design and the type of review. Each team's technical competence encompasses three broad categories: (1) those with broad experience on similar products; (2) those with specialized technical expertise such as in heat transfer, materials, and structural analysis; and (3) those with a functional expertise such as QA, manufacturing, engineering, and product service.

For the ABWR design, the lead design organization prepares the document and circulates it internally for engineering review, approval, and design verification. Evidence of verification is entered into the design records of the responsible design organization. Each document is distributed to the design organizations of the other parties for their review and approval of technical content and design interfaces. All comments resulting from this process are resolved. After resolution of the comments, the design verification is reviewed and, when necessary, updated to ensure that changes did not invalidate the original verification. After final agreement is reached, the document is finalized by the lead design organization, circulated to the other parties for their approval signatures, and then issued. Changes to ABWR documents are handled similarly.

Differences between international and domestic designs are identified in a controlled list for future design action and application.

GE Nuclear Energy's QA organizations are responsible for establishing and implementing the audit program. Audits are performed in accordance with preestablished written checklists by qualified personnel not having direct

responsibilities in the areas being audited. Periodic audits are performed to evaluate all aspects of the QA program including the effectiveness of implementation. The QA program requires the review of audit results by the person having responsibility in the area audited to determine and take corrective action where necessary.

Followup audits are performed to determine if nonconformances and deficiencies have been effectively corrected and the corrective action precludes repetitive occurrences. Audit reviews, which indicate performance trends and the effectiveness of the QA program, are reported to responsible management for review and assessment.

17.1.4 Conclusion

On the basis of its detailed review and evaluation of the QA program description contained in Chapter 17 of the ABWR SSAR, the staff concludes the following:

(1) The organizations and persons performing QA functions have the required independence and authority to effectively carry out the QA program without undue influence from those directly responsible for cost and schedule.

(2) The QA program describes requirements, procedures, and controls that, when properly implemented, comply with the requirements of Appendix B to 10 CFR Part 50 and with the acceptance criteria in Section 17.1 of the Standard Review Plan (NUREG-0800, Rev. 2).

Accordingly, the staff concludes that the description of GE's QA program is in complies with applicable NRC regulations.

17.1.5 Implementation

During its review of the QA program described in the ABWR SSAR, the staff audited the implementation of the program at GE's offices in San Jose, California. The report of this audit is in the Commission's Public Document Room, the Gelman Building, 2120 L Street NW, Washington, D.C. On the basis of the sample

of design activities audited, which included Hitachi and Toshiba documents requested by the staff and translated into English, the auditors concluded that the design QA programs implemented by GE, Hitachi, and Toshiba meet the applicable requirements of Appendix B to 10 CFR Part 50 and are acceptable for designing the ABWR.

TABLE 17.1

Quality Assurance Regulatory Guide Commitments

Number	Applicable Revision	Date
1.8	1	9/75
1.26	3	2/76
1.28*	3	8/85
1.29	3	9/78
1.30	0	8/72
1.37*	0	3/73
1.38*	2	5/77
1.39	2	9/77
1.58*	**	
1.64*	**	
1.74	**	
1.88*	**	
1.94	1	4/76
1.116*	0-R	6/76
1.123*	**	
1.144	**	
1.146*	**	

* NRC accepted the GE Nuclear Energy positions given in the quality assurance topical report, NEDO-11209-04A, Revision 7, May 1987.

** Superseded by Revision 3 of Regulatory Guide 1.28.

18 CONTROL ROOM DESIGN REVIEW

(This section will be provided in a supplement to this SER.)

19 SEVERE ACCIDENT DESIGN CONSIDERATIONS

(This section will be provided in a supplement to this SER.)

APPENDIX A

Advanced Boiling Water Reactor
Licensing Review Bases

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

AUG 7 1987

Project No. 671

Mr. Ricardo Artigas, Manager
Licensing & Consulting Services
General Electric Company
Nuclear Energy Business Operations
175 Curtner Avenue, Mail Code 682
San Jose, California 95125

Dear Mr. Artigas:

SUBJECT: ADVANCED BOILING WATER REACTOR LICENSING REVIEW BASES

As you know, for the past several months the staff has been developing, with coordination and input from GE, certain licensing review bases for the staff review of the forthcoming Advanced Boiling Water Reactor (ABWR) Final Design Approval/Design Certification application. These bases address the review process and selected technical issues. In certain key areas, Commission policies and staff positions are still under development. The Licensing Review Bases represent our understanding of certain approaches which GE has proposed and committed to follow in the ABWR design and license application in order to permit the review to proceed efficiently until final Commission positions and staff requirements are defined and implemented. A copy is enclosed for your information and use. The staff believes that these GE proposals and commitments are adequate to start the review of the ABWR SAR upon submittal.

Please let me know if you have any questions.

Sincerely,

T E Murley

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Enclosure:
As stated

8708140039

Enclosure

GE
Advanced Boiling Water Reactor

Licensing Review Bases

August, 1987

8708140043 (230)

1 INTRODUCTION

GE intends to submit an application for final design approval (FDA) and design certification (DC) for the Advanced Boiling Water Reactor (ABWR). Initial portions of the Safety Analysis Report (SAR) will be submitted beginning in the fall of 1987. Both the NRC staff and GE believe that staff's safety review of the SAR will proceed more smoothly if certain licensing review bases are established before the review starts. These bases are intended to address aspects of the review process and certain technical issues that have caused difficulties in past reviews of standard plant designs.

The Standard Review Plan (SRP) is the basic staff document which will be used in the ABWR review. The Licensing Review Bases addressed in this enclosure provide supplementary guidance on regulatory issues and areas which are either not addressed at all, or not covered in detail, by the SRP. In many cases, these are areas where the staff's positions and requirements are evolving.

These Licensing Review Bases contain no new regulatory requirements. In certain key areas where Commission policies and staff positions are still under development, both the staff and GE have committed to implement acceptance criteria which, if satisfied by the ABWR standard design, would result in a licensable design. Should, however, substantial new information become available that results in new requirements being promulgated by the NRC, they will be addressed during the course of the ABWR review.

The staff supports the efforts of the Department of Energy (DOE) and GE to obtain design certification (DC) of the ABWR. Once the design has been certified, it could be referenced by a number of applicants for use on a number of different sites without further review, except for matters which cannot be reviewed or accepted until a specific facility is constructed. These matters would be specifically identified in the staff Safety Evaluation Report (SER). When an applicant references the pre-approved design, the staff would conduct a compliance review to confirm that the plant was built in accordance with the DC. The design would be certified for the period specified in the Commission's Policy Statement on Standardization, with an option for renewal.

GE has agreed that all Generic and Unresolved Safety Issues relevant to the ABWR will be resolved for the ABWR design before a Final Design Approval (FDA) is issued. After an FDA is issued, new issues will be considered for backfitting under the provisions of 10 CFR 50.109, or under other applicable Commission regulations.

GE is to provide a Safety Analysis Report (SAR) for the entire nuclear island design. The SAR will meet all applicable Commission regulations and contain enough information for the staff to complete its safety review.

1.1 Scope and Content of the ABWR SAR

The scope of the ABWR standard design, as illustrated in Figure 1-1, is a nuclear island. The SAR is to include all of the information necessary for the staff to complete its SER. This includes interfaces (design, construction, testing and operational) between the ABWR standard design and the remainder of plant. GE may later expand its submittal to include some or all of the remainder of plant.

The ABWR standard design employs an advanced boiling water reactor enclosed in a steel-lined reinforced concrete containment vessel integrated with the reactor building which, in turn, forms the secondary containment boundary. The reactor building houses the equipment associated with auxiliary systems (such as emergency core cooling, residual heat removal and reactor water cleanup systems). It also houses the fuel handling and storage and the diesel generators, which have traditionally been housed in separate buildings. A separate control building is located adjacent to the reactor building. The control building includes the control room, change rooms and plant supervisor's office and provides plant access control.

Because GE wishes to obtain an FDA and DC for the ABWR design before any applicant, site, architect/engineer or equipment suppliers are identified, it is necessary that GE provide the necessary level of detailed information to enable the staff to complete its review without preempting competitive bidding on any future project that references the certified design. The technical information for the ABWR standard design portion of the submittal must meet the requirements of 10 CFR 50.34(g) and the guidance in Regulatory Guide 1.70, Revision 3, appropriate to the degree of design available for standard designs. The corresponding contents of the SAR are listed in Section 8.4.1.

Tests, inspections, analyses, and acceptance criteria necessary for an applicant to assure that the designs are properly implemented in the plant will also be defined in the SAR. The applicant will later demonstrate compliance with this design and implementation information.

Section 8.4.1 lists the design documentation that GE intends to submit to support the ABWR standard design. GE does not plan to submit these types of supporting documents for the remainder of plant for the DC effort.

1.2 Scope and Content of Future Applications Referencing the ABWR

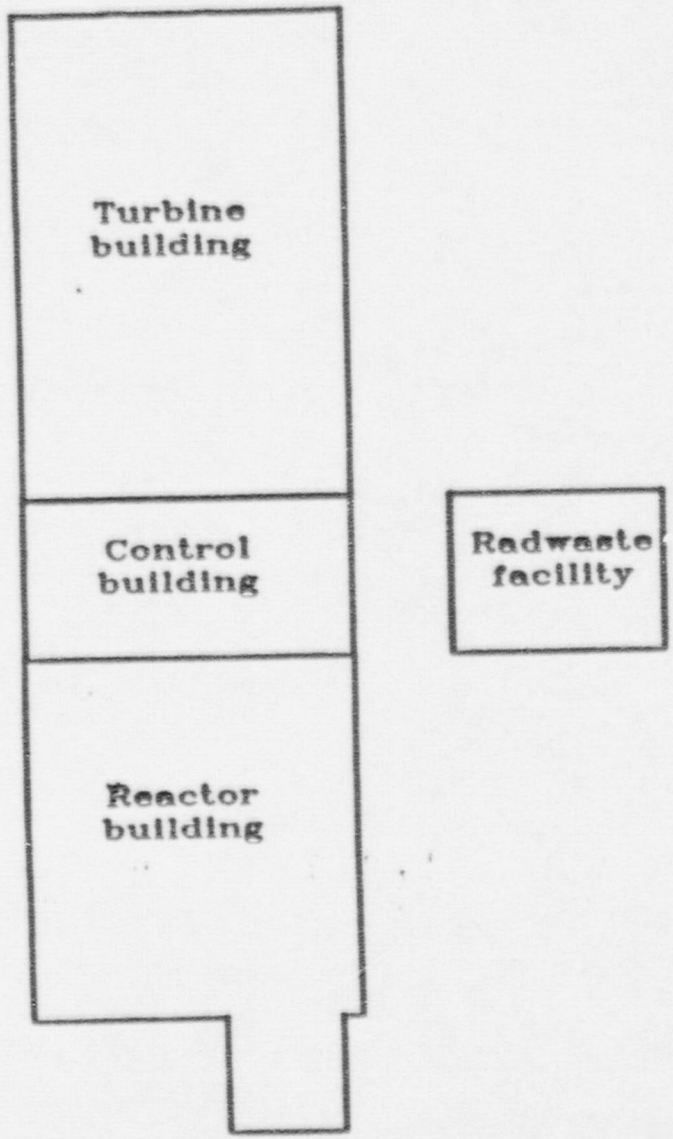
When the approved ABWR standard design is referenced in an application, the staff's review of matters related to the approved design need consider only whether the interface requirements have been satisfied in the referencing application (the applicant's Final Safety Analysis Report (FSAR)). Specifically, for those areas in the remainder of the plant and the site envelope where the ABWR SAR has specified interface requirements, the applicant will have to demonstrate compliance with them. No further review of the referenced design will be required when the site envelope parameters fall within the design envelope and the interface requirements are met.

2 SCHEDULE

The schedule for the FDA review of the ABWR design is shown in Table 2-1. The schedule for the subsequent design certification rulemaking phase depends on, among other factors, the type of rulemaking proceeding selected by the Commission. The range of dates shown is the staff's current best estimate of the design certification duration.

Figure 1-1

Typical ABWR Site Plan



Nuclear Island

- Nuclear Steam Supply System
- Primary Containment
- Secondary Containment
- Emergency Core Cooling Sys.
- Residual Heat Removal Sys.
- Emergency Diesel Generators
- Essential Electrical Power
- Fuel Storage & Handling Equipment
- Control Room
- Plant Access Control
- Emergency Response Center

Remainder of Plant

- Turbine Building
- Radwaste Facility
- Condensate and Diesel Oil Storage

Table 2-1 - ABWR FDA Review Schedule

Review Element	Review Schedule*	Cumulative Elapsed Time, Months**
Chapters 4, 5, 6, 15 (Reactor, Reactor Coolant System, Engineered Safety Features, Accident Analyses)	9/87-3/89	19
Chapters 1, 2, 3, 17 (General Description, Site Characterization, Design of Structures, Components, Equipment, and Systems, QA)	3/88-9/89	25
Chapters 7-9, 11-14, 16 (I&C, Electric Power, Auxiliary Systems, Radioactive Waste, Radiation Protection, Conduct of Operations, Initial Tests and Operation, Technical Specifications)	6/88-12/89	28
Chapters 10, 18 (Steam and Power Conversion, Emergency Planning)	1/89-12/89	28
PRA and Failure Modes & Effects Analysis (FMEA)	1/89-12/89	28
Integrated Review/Final SER	3/89-2/90	30
ACRS Review	9/87-4/90	32
Proposed Decision Date for FDA Design Certification	9/90	37
		49-61

* Time from submittal of chapters to issuance of draft SERs
** From beginning of NRC review

3 CONTENT OF APPLICATION

3.1 Safety Analysis Report Format

The ABWR SAR and all subsequent SAR amendments are to be organized in accordance with Regulatory Guide 1.70, Revision 3, and the SRP in effect on March 30, 1987. The application will include the information specified by 10 CFR 50.33, 50.34, and Appendix O to Part 50. GE will comply with the provisions of 10 CFR 50.34(g)(1)(ii).

3.2 Use of Metric Units

Because the ABWR has been designed for international applications, the SAR may use metric units in describing equipment dimensions and performance. However, the values in the SAR are to have their corresponding English units included, in parentheses, next to the metric values (e.g., maximum fuel cladding temperature during an accident is 1204 °C (2200°F)). GE intends to include a table of conversion factors between metric and English units at the front of each volume of the SAR.

3.3 Data Interchange Format

Because of the increased use of computers in the licensing process, GE plans to provide a copy of the SAR on a diskette suitable for use on an IBM (or compatible) personal computer (except for drawings and graphs that are not amenable to such portrayal). GE also is to provide the requisite number of hard copies of the SAR specified in 10 CFR 50.30(a), (c)(1) and (3).

4 INCORPORATION OF FUTURE ISSUES

As stated in its Severe Accident Policy Statement (see Section 7 below), the Commission expects all new power plant designs to address all Unresolved Safety Issues (USIs) and all medium-and high-priority Generic Safety Issues (GSIs). NUREG-1197, "Advanced Light Water Reactor Program", December 1986, presents these issues and their status as of July 1, 1986. GE is to identify which issues are applicable to the ABWR design and address them. These issues will include both applicable issues identified in NUREG-1197 and any new generic issues raised up to the time of FDA issuance. It is the intention of the staff that there will be no open items regarding the resolution of USIs or GSIs or other plant features for the ABWR at the time of the FDA decision.

Issues introduced after an FDA is issued would be analyzed and resolved in accordance with the backfit requirements of 10 CFR 50.109, except to the extent that the DC rulemaking provides otherwise.

5 STAFF REVIEW PROCEDURES

The staff will follow its review procedures in the SRP, supplemented and modified as follows:

(1) The ABWR SAR is to be submitted chapter by chapter, over a period of about 16 months. Correspondingly, the staff SER will also be issued in draft form, in sections in accordance with the schedule shown in Section 2. The draft SER sections will be made publicly available.

(2) At the completion of the review of the individual SAR chapters, the staff will perform an integrated review of the application. This review will complement the Probabilistic Risk Assessment (PRA) review, in that it will be an overall assessment of the design. The staff will issue a composite final SER in accordance with the schedule described in Section 2.

(3) It will be important to carefully document open or unresolved issues that may be identified early in the review process, but which cannot be resolved until the completion of later chapters. Each draft SER section will contain a description of such issues. In addition, with the submittal of each chapter of the SAR, GE is to provide an updated check-list which identifies outstanding issues and the future chapter(s) in which resolution is anticipated.

(4) Each draft SER will contain a target schedule for closing outstanding SER issues that is compatible with the target FDA decision date.

6 ACRS PARTICIPATION

One step in the design review of a standard plant is the independent review by the Advisory Committee on Reactor Safeguards (ACRS). The ACRS review of the ABWR design certification process started before submittal of the first chapters of the ABWR SAR. An initial briefing of the ACRS by the staff and GE took place early in 1987. Periodic reviews will address the safety aspects of the design on matters selected by the ACRS. The ACRS review is scheduled to continue through April 1990, when the ACRS will be requested to issue a letter report on its review.

The staff will keep the ACRS informed of the progress of the review by forwarding to it copies of the SAR chapters as they are submitted, along with copies of the draft SERs as they are issued. In addition, the staff will meet with the ACRS, as needed, to discuss the draft SERs.

7 SEVERE ACCIDENT POLICY STATEMENT (SAPS)

7.1 Introduction

On August 8, 1985, the Commission issued a Policy Statement on Severe Accidents (50FR32138, "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," and NUREG-1070, "NRC Policy on Future Reactor Designs"). The policy statement provides criteria and procedures for the licensing of new plants, and sets goals and a schedule for the systematic examination of existing plants. The Commission encouraged the development of new designs that might realize safety improvements and stated that it intends to take all reasonable steps to reduce the chances of occurrence of a severe accident and to mitigate the consequences of such an accident, should one occur. The Commission's licensing criteria for new plant designs are specified in the policy statement.

The Commission also recognized the need to provide defense-in-depth by striking a balance between accident prevention and consequence mitigation, through a better understanding of containment performance, with the understanding that new performance criteria for containment systems might need to be established. It also recognized the importance of such potential contributors to severe accident risk as human performance and sabotage, and determined that these issues should be carefully analyzed and considered in the design and operating procedures for a nuclear facility. Specific discussions of each of the policy statement licensing criteria follow.

7.2 Construction Permit/Manufacturing License Rule

GE will comply with all applicable Commission regulations, including those listed in 10 CFR 50.34(f) applicable to the ABWR, except 10 CFR 50.34(f)(2)(i) and 10 CFR 50.34(f)(3)(iv). Any future applicant that references the ABWR design must satisfy 10 CFR 50.34(f)(2)(i) by providing simulators. With regard to 10 CFR 50.34(f)(3)(iv), GE has stated that the ABWR design has specific features that function to mitigate the consequences of severe accidents within the offsite dose objectives discussed in Section 7.5 (Severe Accident Performance Goals). GE intends to provide justification to demonstrate that a dedicated containment penetration is not required in the ABWR design. The staff will consider this justification as part of its review.

7.3 Resolution of USIs and GSIs

See Section 4, above.

7.4 Probabilistic Risk Assessment (PRA)

GE has proposed certain criteria and methodologies relevant to PRAs in the following sections. These criteria and methodologies will be used by the staff as the bases for their review of the ABWR unless new criteria and methodologies are promulgated by the NRC.

7.4.1 Scope

GE has committed to provide a level-3 PRA for the ABWR design, as defined by the "PRA Procedures Guide," NUREG/CR-2300. The level-3 scope PRA includes the following elements:

(1) An analysis of the plant design and operation focused on the accident sequences that could lead to a core melt, their basic causes, and their frequencies

(2) An analysis of the physical processes of the accident sequences and the response of the containment

(3) An analysis of the transport of radionuclides to the environment and an assessment of potential public health consequences

Bounding analyses of external events that can be quantified (e.g., seismic, internal fires, internal floods, tornados) are to be included in this evaluation.

7.4.2 Methodology

The PRA is to be based on a methodology that originated with the approach taken in the Reactor Safety Study (WASH-1400), and that has been developed and systematized through applications in numerous plant-specific studies. The general procedures have been documented in NRC NUREG reports, such as the "PRA Procedures Guide" (NUREG/CR-2300) and the "Probabilistic Safety Analysis Procedures Guide" (NUREG/CR-2815).

GE intends to utilize the IDCOR-developed Modular Accident Analysis Program (MAAP) which has been modified by GE for utilization in BWR analyses. If technical disagreements surface between the NRC methods (such as the Source Term Code Package) and MAAP, specific sensitivity studies will be performed. GE is to use the CRAC-II code or other suitable model acceptable to the staff to compute the potential consequences of fission product releases.

7.4.3 Reference by Utility Applicants

The PRA is intended to be applicable to all sites within the ABWR envelope. It is contemplated that applicants will not have to prepare or submit plant-specific PRAs before an operating license is issued, but that the GE PRA would be updated by the licensee within 2 years after a plant is licensed.

7.5 Severe Accident Performance Goals

This section describes the goals for severe accident performance criteria, to which GE has committed for the ABWR design, consistent with existing staff regulations and policy statements pertaining to severe accidents and defense-in-depth through a balance between accident prevention and consequence mitigation. The staff will utilize these goals as the basis for their review unless new criteria are promulgated by the NRC.

7.5.1 Prevention of Core Damage

GE intends to demonstrate by analyses that the likelihood of core damage will have a mean value of less than one in one hundred thousand reactor years (i.e., 1.0×10^{-5}), including both internal and external events. (The staff will determine the adequacy of this goal and the analyses.)

7.5.2 Mitigation of Core Damage

GE has stated that the ABWR design will provide protection against containment failure if a severe accident occurs and results in core damage. GE intends the containment capabilities to include:

a. Measures to reduce the probability of early containment failure for dominant accident sequences

b. Measures to accomodate hydrogen generated from the reaction of the equivalent of 100% of the zirconium in the active fuel clad, consistent with 10 CFR 50.34(f), as provided for by the Severe Accident Policy Statement

c. Highly reliable heat removal systems to reduce the probability of containment failure by loss of heat removal

d. Reliable means to prevent hydrogen deflagration and detonation, consistent with 10 CFR 50.34(f), as provided for by the Severe Accident Policy Statement.

7.5.3 Offsite Consequences for Severe Accidents

GE has committed to meet the following goals:

(1) The expected mean frequency of occurrence of offsite doses in excess of 25 Rem beyond a half mile radius from the reactor is to be less than once per million reactor years (i.e., 1.0×10^{-6}), considering both internal and external events

(2) The containment design is to assure that the containment conditional failure probability is less than one in ten when weighted over credible core damage sequences

8 ADDITIONAL TECHNICAL ISSUES

8.1 Introduction

The ABWR design will incorporate several features that are novel or which have been used in relatively few other nuclear power plants in the United States. In addition, because standardized plant reviews are conducted before actual facility applications are made, these reviews cannot address every aspect of a facility. This section is intended to address some of the issues which arise from these circumstances and that have caused difficulty in previous standard plant reviews.

8.2 Physical Security

8.2.1 Basis for Requirements

The basis for the requirements will be as defined in 10 CFR 73.55, "Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors Against Radiological Sabotage," and other applicable portions of 10 CFR Part 73.

8.2.2 Acceptance Criteria

The ABWR SAR is to include enough information to demonstrate the existence of adequate physical barriers to protect vital equipment in accordance with 10 CFR 73.55(c), "Physical Barriers," and to identify access control points to all vital areas in accordance with 10 CFR 73.55(d), "Access Requirements."

The ABWR SAR will not provide details but is to identify design requirements to be satisfied by an applicant for the following sections of 10 CFR 73.55 (the applicant must then address all remaining requirements):

(b) Physical Security Organization

(e) Detection Aids

(f) Communication Requirements

(g) Testing and Maintenance

(h) Response Requirements

The design requirements are to include reference to existing NRC documents such as Regulatory Guide 5.44, "Perimeter Intrusion Alarm Systems" and NUREG-0908, "Acceptance Criteria Evaluation of Nuclear Power Reactor Security Plans", as well as to industry standards such as IEEE-692-1986, "IEEE Standard Criteria for Security Systems for Nuclear Power Generating Stations."

8.3 Site Envelope Parameters and Soil-Structure Interaction Analysis

Selected site envelope parameters and methods for soil-structure interaction analysis proposed by GE are provided in Appendix A to this enclosure.

8.4 ABWR Design

8.4.1 Completeness of Design

The ABWR SAR is to provide essentially complete design information. The term "essentially complete" is defined as follows:

(1) The SAR will define the major design components and include the results of sufficient engineering to identify, as appropriate:

a. design basis criteria

b. analysis and design methods

c. functional design and physical arrangement of auxiliary, BOP, and NSSS systems

d. plant physical arrangements sufficient to accommodate system and components

e. functional and/or performance specifications for components and materials sufficiently detailed to become a part of associated procurement specifications

f. acceptance/test requirements

g. risk assessment methodology

(2) Design documentation for systems, structures, and components should include as appropriate:

a. design basis criteria

b. plant general arrangements of structures and components, including piping system layouts

c. process and instrumentation diagrams, electrical system layouts, and major conduit and cable tray layouts

d. control logic diagrams

e. system functional descriptions and supporting studies and analyses

f. component and procurement specifications, including acceptance criteria and test requirements

g. construction and installation specifications, including acceptance criteria and test requirements

h. program for the assurance of quality

i. design-related aspects for the emergency plans

j. supporting design documentation such as site envelope data and calculations sufficient to support the level of design detail noted above

k. design-related aspects of the physical security program

l. an ALARA radiation protection plan

m. accident analyses

n. Technical Specifications

o. risk analysis

In the limited cases where design information is not available, GE is to provide information on methods, procedures, and performance criteria. GE also is to define those related tests, inspections, analyses, and acceptance criteria that are necessary to assure that the designs are properly implemented in the plant. These tests, inspections, analyses, and acceptance criteria are intended to be implemented and verified in a series of reviews by the applicant during construction and pre-operation. The staff will monitor the performance of these reviews and implementation of the design through its inspection program.

The degree of design detail necessary for providing an essentially complete design is to be that detail that is suitable for obtaining specific equipment or construction bids and to demonstrate conformance to the design safety limits and criteria.

8.4.2 Program for the Assurance of Quality in Design

The design process and resultant design documents must meet the quality assurance (QA) requirements delineated in Appendix B of 10 CFR Part 50, as addressed in Section 17.1 of the SRP. GE must submit justification, acceptable to the staff, for any deviations from Appendix B.

8.5 Interface Assumptions Affecting Safety Determinations for the Nuclear Island

The nuclear island scope of the ABWR reduces the number of interfaces between the nuclear island and the remainder of the plant. GE is to provide a list of the assumptions relied upon to make safety determinations for the nuclear island design. This listing is to identify the nuclear system, the instrumentation and control requirements, reliability assumptions and specific performance criteria. GE is to use the results of the PRA to indicate which interfaces are particularly sensitive to deviations.

8.6 Instrumentation and Controls

GE has committed to use standards and criteria and provide information pertaining to instrumentation and controls for the ABWR as discussed in Appendix B to this enclosure.

8.7 Water Chemistry Guidelines

The maintenance of proper water chemistry in BWR cooling systems is essential to the prevention of stress corrosion cracking of austenitic stainless steel piping and to the minimization of plant radiation levels due to activated corrosion products. GE has committed to using at least the following documents in this area:

(1) EPRI NP-3589-SR-LD, "BWR Water Chemistry Guidelines," April 1985

(2) EPRI NP-4500-SR-LD, "Guidelines for Permanent Hydrogen Water Chemistry Installations," March 1986; revised, November 1986

(3) EPRI NP-4474, "BWR Radiation-Field Control Using Zinc Injection Passivation," March 1986

8.8 Maintenance and Surveillance

GE is to provide in the SAR the reliability and maintenance criteria that a future applicant must satisfy to ensure that the safety of the as-built facility will continue to be accurately described by the certified design. The SAR is to include the key assumptions of the PRA and other PRA licensing commitments.

8.9 MSIV Allowable Leakage and Related Dose Calculations

GE has committed to an ABWR standard design that will provide a non-safety related main steam isolation valve (MSIV) leakage processing pathway consistent with those evaluated in NUREG-1169, "Resolution of Generic Issue C-8," August 1986. The allowable MSIV leakage is to be determined based on the calculated total dose (using methodologies consistent with NUREG-1169) from all leakage sources and the exposure guidelines of 10 CFR 100.11. In addition, leakage for the final installed MSIV test is to be less than 50% of the value allowed to account for equipment degradation during the design lifetime.

8.10 Safety Goal Policy Statement

On August 4, 1986, the Commission published a policy statement on "Safety Goals for the Operation of Nuclear Power Plants" (51 FR 28044). This policy statement focuses on the risks to the public from nuclear power plant operations. Its objective is to establish goals that broadly define an acceptable level of radiological risk.

Although the implementation requirements for the Safety Goal Policy Statement are still being developed by the staff, GE has committed to severe accident performance standards and criteria that are intended to assure compliance with those eventual requirements.

9 FINAL DESIGN APPROVAL

The staff may issue an FDA after it and the ACRS complete their reviews of the final design. The FDA means that an entire nuclear power plant design or major portion thereof is acceptable for incorporation by reference in individual applications for construction permits, operating licenses, and manufacturing licenses. The staff and the ACRS intend to use and rely on the approved final design in their reviews of those applications. However, an approved final design is subject to litigation in individual licensing proceedings on those applications. An FDA is a prerequisite for a design certification.

10 DESIGN CERTIFICATION

10.1 Introduction

The Commission currently is considering staff-proposed revisions to its 1978 policy statement on standardization of nuclear power plant designs. The Commission also is developing proposed regulations that will address licensing reform and standardization and provide a regulatory framework for implementation of the standardization policy, including Commission certification of standard designs by rulemaking. Since design certification is the ultimate goal of the ABWR program, and since the focus of the proposed policy statement and regulations is reference system design certification, the essence of these proposals, and GE's commitment to them, is summarized here. It should be noted, however, that the Commission has not yet acted on these proposals and that they are subject to change.

The staff-proposed revisions to the policy statement encourage the use of standard plant designs in all future license applications. The staff believes that the use of standard plant designs can benefit public health and safety by:

(1) Concentrating the resources of designers, engineers, and vendors on particular approaches

(2) Stimulating standardized programs of construction practice and quality assurance

(3) Improving the training of personnel

(4) Fostering more effective maintenance and improved operation

The staff believes that the use of such standardized designs can also permit more effective and efficient licensing and inspection by the NRC.

10.2 Design Certification Concept

The design certification concept, as described in the staff's proposed standardization policy statement, provides for certifying a reference system design (such as the ABWR) through rulemaking. In this process, the Commission would certify a design after the staff issues an FDA and a rulemaking proceeding is completed. The design certification means that the portions of the nuclear power plant design that have been reviewed are acceptable for incorporation by reference in an individual license application. The conclusions of the certification rulemaking would be used and relied on by the staff, the ACRS, the hearing boards, and the Commission in their reviews of applications that reference the design. The certified design would not be subject to litigation in individual licensing proceedings, except as provided in 10 CFR 2.758.

Under the staff proposal, the Commission could certify the standardized ABWR design for referencing by applicants for a period of 10 years. Renewal of the design certification could be granted for an additional period of up to ten years unless the Commission found that the design would not comply with the Commission's then-current regulations. Applicants could reference the certified ABWR design in applications for CPs and OLs docketed during the period beginning with the docketing date of the FDA application and ending

at the expiration date of the design certification. However, no CP or OL could be issued for an application referencing the ABWR design until the FDA is issued.

10.3 Completeness of Scope and Design Detail

The ABWR application for design certification is to include a plant design that is essentially complete in both scope and level of detail. The scope of design developed to support the design certification process is addressed in Section 8.4.

The ABWR application for design certification also is to demonstrate compliance with the licensing criteria for new plant designs set forth in the Commission's Severe Accident Policy Statement (Section 7). The ABWR SAR is to address the tests, analyses, and inspections that are necessary to provide reasonable assurance that the plant will be built and operated within the specifications of the certified design. For those individual aspects of the design where safety-related structures and components differ from those of existing designs, empirical information is to be included as part of the application for design certification.

The ABWR SAR is to include information that will permit construction verification and compliance. This will permit reviews during the construction and startup phases of the plant and will eliminate the need for further design reviews on those portions of the plant that have been certified, except to verify that interface requirements have been met.

10.4 Changes to Approved and Certified Designs

The staff believes that standardization will be best achieved if changes to approved or certified designs are kept to a minimum. Nevertheless, there are situations in which changes may be needed or desirable. It is the staff's intent that after issuance of the design certification, the Commission would require backfitting only when it determines, using the standards in 10 CFR 50.109 and the results of the DC rulemaking, that a substantial increase in the overall protection of the public health and safety would result.

GE may request modifications to an approved or certified design by applying for an amendment to the design approval or certification in accordance with the proposed regulations.

10.5 Rulemaking

Appendix O to 10 CFR 50 provides the opportunity for the Commission to approve the ABWR reference system design in a rulemaking proceeding. The regulations that are currently under development will specify the procedures to be used for the rulemaking. In general, however, upon receipt of a request from GE, a notice would be published in the Federal Register announcing the request for a design certification for the ABWR. The notice would set out the matters at issue, as specifically as possible, and the possible hearing procedures that could be used if the Commission decided to hold a hearing, and would request that all persons wishing to participate in a hearing notify the Commission within a stated period of time. As a condition to participating in a hearing, however, intervenors could be required to state the issues they wish to have considered at the hearing and to commit to providing expert testimony

on those issues. Written comments could be invited from those not intending to participate in the hearings. As a result of responses to the notice, or on its own initiative, the Commission could then hold hearings on the proposed rulemaking.

If a hearing were held, the Commission could use a number of formats, from the simple hearing and recording of testimony to interchanges among the parties and a limited right of cross-examination. Because such rulemaking procedures go beyond the notice-and-comment requirements for rulemaking, the Commission has broad discretion to establish hearing procedures best suited to the matters at issue.

After any hearing, the Commission could review the complete record of the rulemaking, including both the hearing record and any other written comments. The notice of final rulemaking would have to include responses to written comments and the resolution of issues considered at a hearing.

The views of the ACRS would be sought and considered. The ACRS would review the design before the rulemaking, and the results of the ACRS review would be made available when the proposed rule is announced.

10.6 Renewal of Certifications

Under the staff's proposal, the Commission could certify the standardized ABWR design for referencing by applicants for a period of 10 years. Additionally, before the expiration of the design certification, GE could apply for certification renewal. The design certification could be renewed for an additional 10-year period, provided the design complies with the Commission's regulations in effect at the time of the renewal application.

APPENDIX A

SITE ENVELOPE PARAMETERS AND SOIL-STRUCTURE INTERACTION ANALYSIS

1 SITE ENVELOPE PARAMETERS

This Appendix addresses some of the important site envelope parameters proposed by GE for the ABWR. Additional parameters are to be provided in the SAR. All of the proposed parameters and analyses will be considered by the staff during the review.

Utility applicants must verify that their proposed facilities lie within the site envelope parameters assumed by GE in its safety analyses and approved by the staff. No further analysis or actions will be required by an applicant when the site-specific parameters fall within the design envelope. If site or interface parameters fall outside the design envelope, utility applicants must provide justification for the deviations.

2 SOIL-STRUCTURE INTERACTION (SSI) ANALYSIS

2.1 Scope

GE will perform soil-structure interaction (SSI) analyses of the reactor building design for a range of site conditions within the site envelope design parameters defined in Table A-1. GE will satisfy the staff acceptance criteria for the design ground motion and SSI analysis methods as specified in Revision 2 of the SRP, Sections 2.5.2, 3.7.1, and 3.7.2.

2.2 Methodology

GE intends to employ the state-of-the-art computer program, SASSI (System for Analysis of Soil-Structure Interaction) code. SASSI is a linear analysis program using the finite element approach. Solutions for a complete soil-structure system are sought in the frequency domain employing the complex response technique. Problem formulations are based on the flexible volume substructuring method in that the complete soil-structure system is partitioned into the foundation and the structure.

In compliance with the duality (finite element and half-space) requirement, limited cases are to also be analyzed using the CLASSI/ASD computer code, which is an improved version of the CLASSI family of computer codes. CLASSI/ASD is a linear analysis program using the structure approach based upon continuum mechanics for half-space.

Table A-1
Envelope Of Selected Plant Site Design Parameters
Applicable To ABWR

MAXIMUM GROUND WATER LEVEL: 2 feet below grade

PRECIPITATION (for roof design):

Maximum rainfall rate: 10 in/hr

Maximum snow load: 50 lb/sq ft

DESIGN TEMPERATURES:

Ambient:

1% Exceedance Values
Maximum: 100°F dry bulb /77°F coincident wet bulb
Minimum: -10°F

0% Exceedance Values (historical limit)
Maximum: 115°F dry bulb /82°F coincident wet bulb
Minimum: -40°F

Peak Emergency Cooling Water Inlet: 95°F

Condenser Cooling Water Inlet: 100°F

SEISMOLOGY:

SSE PGA: 0.30g*

SSE Response Spectra: per Regulatory Guide 1.60

SSE Time History: Envelope SSE Response Spectra

EXTREME WIND: Basic Wind Speed: 110 mph** / 130 mph***

SOIL PROPERTIES:****

Minimum bearing capacity (demand): 15 ksf

Minimum shear wave velocity: 1000 fps

Liquification potential: None at plant site resulting from OBE and SSE

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* Free-field, at plant grade elevation.

** 50-year recurrence interval; value to be used for design of non-safety-related structures only.

*** 100-year recurrence interval; value to be used for design of safety-related structures only.

**** Values of bearing capacity and shear wave velocity are included in this table to ensure wide application of a standard mat-type foundation design. The design must be evaluated parametrically against ranges of possible soil properties to verify wide application.

2.3 Key Design Parameters

(1) Design Ground Motion

For the purpose of standard plant design, GE plans to use a value of 0.3g for the peak ground acceleration (PGA) for the Safe Shutdown Earthquake (SSE) in accordance with Table A-1.

(2) Control Motion Location

The control motion (or design ground motion) is to be defined at the finished grade in the free-field. This is consistent with Revision 2 of the SRP, Section 3.7.1, and with the recommendations in NUREG/CP-0054, "Proceedings of the Workshop on Soil-Structure Interaction", June 1986, for the application of ground motion defined by Regulatory Guide 1.60 for standard design ground spectra.

(3) Design Ground Spectra

The design ground spectra are to be the Regulatory Guide 1.60 standard design ground spectra, normalized to the design peak ground acceleration.

(4) Design Time History

A single set of three artificial time histories (two horizontal components and one vertical component) are to be considered. They are to satisfy the spectra enveloping requirement of the Regulatory Guide 1.60 spectra. The power spectral density (PSD) function of the two horizontal components are to be calculated and compared with the target PSD specified in Revision 2 of SRP, Section 3.7.1.

(5) Soil Conditions

To support the ABWR all-soil envelope design concept, GE plans to analyze a reasonably large number of cases by considering the variations of soil-site conditions within the site envelopes of Table A-1.

The site conditions are to be selected based on those considered in the GESSAR II design and adjusted to accommodate the ABWR design. These site conditions cover a wide range of soil deposit depths, shear wave velocities, and water table locations.

(6) Strain-Dependent Soil Properties

GE plans to use strain-dependent soil properties for shear modulus and material damping, as defined in the GESSAR II design. In accordance with NUREG/CR-1161, "Recommended Revisions to NRC Seismic Design Criteria", May 1980, the strain-compatible shear modulus will be limited to 40% of its low-strain value and material damping will be limited to 15% of critical. The effects of pore pressure are to be taken into account by varying the water table location.

(7) Floor Response Spectra Peak Broadening

The floor response spectra to be generated for subsystem seismic design are to be the enveloped spectra for a wide range of potential site conditions. The uncertainties associated with soil properties, therefore, will be automatically accounted for in the enveloped spectra. The only other sources of uncertainty that may exist in the calculated floor response spectra are those associated with modelling approximations and structural material properties. To account for those uncertainties that may result in variations in structural frequencies, GE considers it sufficient to broaden the peaks at the structural frequencies of the enveloped computed floor response spectra by +10%. This is consistent with the GESSAR II design.

APPENDIX B

INSTRUMENTATION AND CONTROLS

1 INTRODUCTION

The instrumentation and control (I&C) systems of the ABWR are to use state-of-the-art fiber optics, multiplexing, and computer controls. Staff guidance in this area has not been developed, however GE has committed to the standards and criteria currently specified in the SRP, and use of the documents and criteria identified below. GE is to provide the information listed below so the staff can determine the acceptability of the ABWR I&C systems.

In lieu of actual test or qualification reports for equipment that will be selected later by GE or a utility applicant, GE is to provide a detailed description of the testing or qualification standards to be used to assure that the equipment that is ultimately selected will perform as intended.

2 MULTIPLEXING SYSTEMS

In the SAR, GE is to:

(1) Provide a complete list of components (pumps, valves, etc.) whose actuation, interlock, or status indication is dependent on the proper operation of each Class 1E multiplexer.

(2) For the components cited above, describe the means of remote or local control (other than by cutting wires or jumpering) that may be employed should the multiplexer fail.

(3) Describe the multiplexer pre-operational test program.

(4) Describe the test and/or hardware features employed to demonstrate fault tolerance to electro-magnetic interference.

(5) Describe the interconnection, if any, of any Class 1E multiplexer to non-Class 1E devices such as the plant computer.

(6) Describe the online test and/or diagnostic features that may be employed, including any operator alarms/indicators and their locations.

(7) Describe the multiplexer power sources.

(8) Describe the dynamic response of the multiplexers to momentary interruptions of AC power.

(9) Describe the applicability of the plant Technical Specifications to multiplexer operability.

(10) Describe the hardware architecture of all multiplexer units.

(11) Describe the "firmware" architecture.

(12) Provide an explicit discussion of how the systems conform to the provisions of IEEE-279, Section 4.17.

(13) Provide an explicit discussion of how the systems conform to IEEE-279, paragraph 4.7.2, as supplemented by Regulatory Guide 1.75 and IEEE-384.

(14) Provide confirmation that system level failures of any multiplexer system detected by automated diagnostic techniques are indicated to the operators consistent with Regulatory Guide 1.47.

(15) Provide an explicit discussion of the susceptibility of the multiplexer systems to electromagnetic interference.

3 ELECTRICAL ISOLATORS

GE has committed to provide the following on isolation devices:

(1) For each type of device used to accomplish electrical isolation, a description of the testing to be performed to demonstrate that the device is acceptable for its application(s). The test configuration and how the maximum credible faults applied to the devices will be included in the description.

(2) Identification of the data that will be used to verify that the maximum credible faults applied during the test are the maximum voltage/current to which the device could be exposed, and to define how the maximum voltage/current is determined.

(3) Identification of the data that will be used to verify that the maximum credible fault is applied to the output of the device in the transverse mode (between signal and return) and other faults are considered (i.e., open and short circuits).

(4) A definition of the pass/fail acceptance criteria for each type of device.

(5) A commitment that the isolation devices will comply with all environmental qualification and seismic qualification requirements.

(6) A description of the measures taken to protect the safety systems from electrical interference (i.e., electrostatic coupling, EMI, common mode, and crosstalk) that may be generated.

(7) Information to verify that the Class 1E isolation devices are powered from a Class 1E power source(s).

(8) A comparison of the design with the guidance in NUREG/CR-3453/EGG-2444, "Electronic Isolators Used in Safety Systems of U.S. Nuclear Power Plants," March 1986.

(9) A comparison of the design with the guidance in draft Regulatory Guide EE502-4, "Criteria for Electrical Isolation Devices Used in Safety Systems for Nuclear Power Plants".

4 FIBER OPTIC CABLE

The staff is working with EG&G to develop comprehensive guidance on this subject. The guidance will be based on the existing IEEE cable standards, such as IEEE-323 and IEEE-384, on the ANSI standards for fiber optic cables which are listed at the end of this Appendix), and the results of the EG&G work.

5 PROGRAMMABLE DIGITAL COMPUTER SOFTWARE

As a starting point, the following documentation is to be used by GE in the design and by the staff in its review:

(1) ANSI/IEEE-ANS-7.4.3.2, "Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations," 1982

(2) Regulatory Guide 1.152, "Criteria for Programmable Digital Computer System Software in Safety-Related Systems of Nuclear Power Plants," November 1985

(3) NUREG-0308, "Safety Evaluation Report - Arkansas Nuclear 1, Unit 2," November 1977

(4) NUREG-0493, "A Defense-in-Depth and Diversity Assessment of the RESAR-414 Integrated Protection System," May 1985

(5) NUREG-0491, "Safety Evaluation Report of RESAR-414," February 1979

6 PROGRAMMABLE DIGITAL COMPUTER HARDWARE

As a starting point, the following documentation is to be used by GE in the design and by the staff in its review:

(1) IEEE 603, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," 1980

(2) NUREG-0308, "Safety Evaluation Report - Arkansas Nuclear 1, Unit 2," November 1977

(3) Regulatory Guide 1.153, "Criteria for Power, Instrumentation, and Control Portions of Safety Systems," December 1985

(4) NUREG-0493, "A Defense-in-Depth and Diversity Assessment of the RESAR-414 Integrated Protection System," May 1985

(5) NUREG-0491, "Safety Evaluation Report of RESAR-414," February 1979

ANSI Standards on Fiber Optics

FIBER OPTICS

Accelerated Oxygen Aging, ANSI/EIA RS-455-101-1984, $7.00

Acceleration Testing of Components and Assemblies, ANSI/EIA 455-184-1984, $5.00

•Air Leakage Testing of Fiber Optic Component Seals, ANSI/EIA 455-234-1985, $5.00

Altitude Immersion, ANSI/EIA RS-455-15-1983, $5.00

Ambient Light Susceptibility, ANSI/EIA RS-455-22-1983, $7.00

•Bend Test for Fiber Optic Cable Twist, ANSI/EIA 455-91-1985, A

Buffered Fiber Bend Test, ANSI/EIA RS-455-103-1984, $5.00

Cable to Interconnecting Device Axial Compressive Loading, ANSI/EIA RS-455-83-1982, $5.00

•Cables, Generic Specification for Fiber Optic, ANSI/EIA RS-472-1985, $18.00

•Communication Cables for Indoor Use, Sectional Specification for Fiber Optic, ANSI/EIA 472-C-1985, $8.00

•Communication Cables for Outside Aerial Use, Sectional Specification for Fiber Optic, ANSI/EIA 472-A-1985, $8.00

•Communication Cables for Outside Telephone Plant Use, Sectional Specifications for Fiber Optic, ANSI/EIA 472-D-1985, $8.00

•Communication Cables for Underground and Buried Use, Sectional Specifications for Fiber Optic, ANSI/EIA 472-B-1985, $8.00

•Compound Flow (Drip) Test for Filled Fiber Optic Cable, ANSI/EIA 455-81-1985, $5.00

•Compressive Loading Resistance of Fiber Optic Cables, ANSI/EIA RS-455-41-1985, A

Connector Terminology, Fiber Optic, ANSI/EIA RS-440-1978, $5.00

•Connector/Component Temperature Life, ANSI/EIA 455-4A-1985, $5.00

Connectors, Generic Specification for, ANSI/EIA RS-475-1982, $13.00

Core Diameter Measurement of Graded Index Optical Fibers, ANSI/EIA RS-455-58-1984, $7.00

•Crush Resistance of Fiber Optic Interconnecting Devices, ANSI/EIA 455-26A-1985, $5.00

•Definitions of Terms Relating to Fiber Optics, ANSI/IEEE 812-1984, $7.00

Evaluating Fungus Resistance of Optical Waveguide Fibers, ANSI/EIA RS-455-56-1983, $8.00

External Fiber Optic Cable External Freezing Test, ANSI/EIA RS-455-88-1983, $5.00

Fiber Geometry of Optical Waveguide Fibers, Microscopic Method for Measuring, ANSI/EIA RS-455-45-1984, $8.00

Fiber Optic Cable Bend Test, Low and High Temperature, ANSI/EIA RS-455-37-1983, $5.00

Fiber Optic Cable Twist Test, ANSI/EIA RS-455-85-1984, $7.00

Fiber Optic Cable Wicking Test, ANSI/EIA RS-455-39-1983, $5.00

Fiber Optic Circuit Discontinuities, ANSI/EIA RS-455-32-1983, $5.00

Fiber Optic Connector Dust (Fine Sand) Test, ANSI/EIA RS-455-35-1983, $5.00

Fiber Optic Shock Test (Specified Pulse), ANSI/EIA RS-455-14-1983, $9.00

Fluid Immersion Test for Fiber Optic Cables, ANSI/EIA RS-455-60-1982, $8.00

Gas Flame Test for Fiber Optic Cable, ANSI/EIA RS-455-99-1982, $5.00

Gas Leakage Test for Fiber Optic Cable, ANSI/EIA RS-455-100-1984, $7.00

Impact Testing Measurements for Devices, ANSI/EIA RS-455-24-1984, $5.00

•Interconnection Device Insertion Loss Test, ANSI/EIA 455-34-1985, $6.00

Jacket Elongation and Tensile Strength for Fiber Optic Cable, ANSI/EIA RS-455-89-1983, $5.00

Jacket Self-Adhesion (Blocking) Test for Fiber Optic Cable, ANSI/EIA RS-455-84-1984, $5.00

Jacket Shrinkage Test for Fiber Optic Cable, ANSI/EIA RS-455-86-1983, $5.00

Knot Test for Fiber Optic Cable, ANSI/EIA RS-455-87-1983, $8.00

Light Launch Conditions for Long Length Graded-Index Optical Fiber Spectral Attenuation Measurements, ANSI/EIA RS-455-50-1983, $8.00

Mating Durability for Fiber Optic Interconnecting Devices, ANSI/EIA 455-21-1984, $5.00

Measurement of Change in Optical Transmittance, ANSI/EIA RS-455-20-1983, $8.00

Measuring Relative Abrasion Resistance of Optical Waveguide Coatings and Buffers, ANSI/EIA RS-455-66-1984, $7.00

On-Line Diameter Measurement of Optical Waveguides, ANSI/EIA RS-455-48-1983, $5.00

•Optical Crosstalk in Fiber Optic Components, ANSI/EIA 455-42-1985, $6.00

Optical Waveguide Fiber Material Classes and Preferred Sizes, ANSI/EIA RS-458-A-1984, $7.00

•Optical Waveguide Fibers, Generic Specification for, ANSI/EIA 492-1985, $14.00

Output Far-Field Radiation Pattern Measurement, ANSI/EIA RS-455-47-1983, $10.00

Output Near-Field Radiation Pattern Measurement of Optical Waveguide Fibers, ANSI/EIA 455-43-1984, $5.00

Procedure to Measure Nuclear Radiation Effects in Fiber Optic Components, ANSI/EIA RS-445-45-1982, $8.00

•Distortion Measurement of Multimode Glass Optical Fiber Information Transmission Capacity, ANSI/EIA RS-455-51-1983, $10.00

Refractive Index Profile, Refracted Ray Method, ANSI/EIA RS-455-46-1984, $9.00

Salt Spray (Corrosion) Test for Components, ANSI/EIA RS-455-16-1984, $7.00

Spectral Attenuation Measurement for Long-Length, Graded-Index Optical Fibers, ANSI/EIA RS-455-46-1983, $5.00

Temperature Dependence of Attenuation for Optical Waveguide Fibers, Method for Measuring, ANSI/EIA RS-455-52-1982, $8.00

Terminal Devices, Generic Specification for, ANSI/EIA RS-509-1984, $10.00

Test Procedure 3, ANSI/EIA RS-455-1-1981, $10.00

Test Procedures for Fiber Optic Fibers, Cables, Transducers, Connecting and Terminating Devices, ANSI/EIA RS-455-1980, $13.00

Test Procedures 12, 27, 28, and 29, ANSI/EIA RS-455-4-1981, $14.00

•Test Procedures 17, 25, and 31, ANSI/EIA RS-455-3-1980(R1985), $13.00

•Test Procedures 26 (Crush Resistance), ANSI/EIA RS-455-26A-1985, $5.00

•Test Procedures 30, 33, 36, 54, 55, and 82, ANSI/EIA RS-455-5-1982, $14.00

Torsion Test for Optical Waveguide Fibers, ANSI/EIA RS-455-63-1984, $7.00

Visual and Mechanical Inspection of Fibers, Cables, Connectors, and/or Other Fiber Optic Devices, ANSI/EIA RS-455-13-1984, $5.00