May 1, 1990

Docket No. STN 50-605

Patrick W. Marriott, Manager
Licensing & Consulting Services
GE Nuclear Energy
General Electric Company
175 Curtner Avenue
San Jose, California 95125

Dear Mr. Marriott:

SUBJECT: RESOLUTION OF OUTSTANDING SAFETY EVALUATION ISSUES RELATING TO THE GENERAL ELECTRIC COMPANY APPLICATION FOR CERTIFICATION OF THE ABWR DESIGN

Enclosed are copies of Preliminary Draft Safety Evaluation (PDSER) sections relating to the staff's review of your application for certification of the Advanced Boiling Water Reactor Design. In these PDSER sections we have identified a need for additional information in the form of outstanding issues. In order for us to maintain the ABWR review schedule, we request that you provide a schedule that is consistent with resolving the identified outstanding issues by the end of May 1990. If you have any concerns regarding this request please call me on (301) 492-1104.

Sincerely,

|s|

Dino C. Scaletti, Project Manager
Standardization and Life
Extension Project Directorate
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
As stated

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

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CHAPTER 3
PLANT SYSTEMS BRANCH

3.5.1.1 Internally Generated Missiles (Outside Containment)

The design of the facility for protecting structures, systems and components important to safety against internally generated missiles outside containment was reviewed in accordance with SRP Section 3.5.1.1. Specifically, the review included the missile protection design features for the structures, systems, and components whose failure could prevent safe shutdown of the facility or result in significant uncontrolled release of radioactivity. 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 provided in the "Review Procedures" portion of the SRP section. The acceptance criteria formed the basis for the staff's evaluation of the design of the facility for protection against internally generated missiles with respect to the applicable regulations of 10 CFR Part 50. The SRP acceptance criteria require the design to meet GDC 4, "Environmental and Dynamic Effects Design Bases," as it relates to protecting the structures, systems, and components outside containment against the effects of missiles that can be internally generated during facility operation. The acceptance criteria for the design of the facility for missile protection include meeting the guidelines of Regulatory Guide 115, "Protection Against Low-Trajectory Turbine Missiles." The staff's review of the internally generated missiles did not include turbine missiles since these are discussed separately in Section 3.5.1.3 of this SER. The review included all areas outside containment that were within the scope of the ABWR.

GE evaluated potential internally generated missiles resulting from plant equipment failures within the nuclear island and located outside containment. Potential missiles identified by this analysis were categorized into two groups:

(1) Potential missiles that could result from the failure of rotating diesel generators and compressors, and

(2) Pressurized high energy fluid system components considered potential missile sources including valve bonnets, stems, pressure vessels, thermowells, retaining bolts and blowout panels. Probability calculations were also performed for certain rotating equipment and pressurized components to identify qualifying missiles. Piping failures were not included as sources of potential internally generated missiles since the whipping section remains attached to the remainder of the pipe and the dynamic effects associated with this type of break are addressed in Section 3.6.

The primary means of protecting structures, systems, and components important to safety from internally generated missiles is to provide design features to limit missile generation. In addition, protection from internally generated missiles is provided in one or more of the following ways. The systems and components to be protected are located in individual missile-proof structures or localized protective shields and barriers are used. Potential missile sources are oriented to prevent unacceptable consequences due to missile generation. Physical separation of redundant components is used to prevent damage from a missile. The adequacy of structures, shields and barriers provided for missile protection is evaluated in Section 3.5.3 of this SER.

GE stated that rotating equipment such as pumps and fans have synchronous motors. However, the staff believes that some of them may be induction motors. Since synchronous motor speed is related to the line frequency which is fairly stable, the pumps and fans are unlikely to attain an overspeed condition. Fan blade casings are designed with sufficient thickness such that a fan blade breaking off at rated speed will not penetrate the fan casing. Valve bonnets have sufficient design safety factors (based on the ultimate strength of the materials) to prevent them from becoming credible missiles. Valve stems have design features such as stem threads and backseats to prevent their ejection. Nuts, bolts, nut and bolt combinations and nut and stud combinations have insufficient stored energy to require missile protection analysis. GE analyzed the thermowells and concluded that their maximum ejection velocity was insufficient to cause damage to safety-related systems. Blowout panels are restrained by hinges to prevent the panels from becoming credible missiles. Air bottles are located, oriented and restrained in a manner sufficient to prevent them from becoming missiles.

Based on its review of the above features, discussed in the ABWR SSAR Section 3.5.1.1, the staff agrees with the above findings, subject to GE's clarification of the use of induction motors for some applicable rotating equipment. However, the staff will review plant-specific details of air bottle design for each plant that references the ABWR design. Regarding the design features provided for compliance with GDC 4, GE has not provided sufficient information to determine that for all safety- related systems, adequate physical separation exists between redundant trains that do not have missile-proof barriers. GE has not described the means by which safety-related systems will be protected from missiles generated by nonsafety-related components and the means by which stored spent fuel will be protected from internally generated missiles. Additionally, GE has not provided a listing of all local shields and barriers. Until the above information is provided, the staff cannot conclude that the ABWR design meets the requirements of GDC 4 regarding protection from internally generated missiles outside containment. For the structures, systems, and components outside the containment that are not within the ABWR design scope, the staff will evaluate missile protection individually for each application that references the ABWR design.

3.5.1.2 Internally Generated Missiles (Inside Containment)

The design of the facility for protecting structures, systems, and components important to safety against internally generated missiles inside containment was reviewed in accordance with SRP Section 3.5.1.2. Specifically, the review included missile protection design features for the structures, systems, and components whose failure could prevent safe shutdown of the facility or result in significant uncontrolled release of radioactivity. 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 provided in the "Review Procedures" portion of the SRP section. The acceptance criteria formed the basis for the staff's evaluation of the design of the facility for protection against internally generated missiles with respect to the applicable regulations of 10 CFR Part 50. The SRP acceptance criteria require the design to meet GDC 4 as it relates to protecting the structures, systems, and components inside containment against the effects of missiles that can be internally generated during facility operation.

GE evaluated potential internally generated missiles resulting from plant equipment and component failures within the containment structure. The potential missiles identified by this analysis were categorized into three groups: those generated by rotating equipment (e.g., pump impellers, compressors, and fan blades); missiles generated by pressurized components (e.g., valve bonnets, thermowells, nuts, bolts, studs, valve stems, and accumulators); and gravitational missiles.

GE's analysis of rotating equipment failures indicates that equipment design prevents such equipment from becoming sources of potential missiles. Pumps are unlikely to achieve an overspeed condition. For example, all the reactor internal pumps and motors are designed for deflection and high cycle fatigue to provide sufficient safety factors to prevent fracture and bursting of the pump impellers and motor rotors due to backward rotation of the pumps during a LOCA or normal operation with one idle pump. Additionally, the pumps have mechanical backstops which function like a one-way clutch and anti-rotation devices at the bottom of the pump motors to prevent reverse overspeed and consequent generation of missiles. GE has also performed an impeller missile study to demonstrate that even if missiles are generated from impellers despite the above design features, they will not penetrate the reactor pressure vessel or the shroud walls.

Pressurized components and equipment such as valve bonnets, valve stems, nuts, bolts, nut, and bolt combinations, nut and stud combinations, thermowells, and blow-out panels inside containment are not considered credible missiles for the same reasons (i.e., design features or insufficient stored energy) as stated in Section 3.5.1.1 of this SER. Fine motion control rod drive mechanisms under the reactor vessel are not credible missiles since the housings are designed to prevent any significant nuclear transient in the event of a drive housing break. Specifically, the pressure boundary containing the fine motion control rod drive mechanisms, including the bolted flange connections, are stressed below the ASME Code limits and meets all code requirements. Also, to prevent control rod drop accidents, internal restraints are provided to support the fine motion control rod drive housing in the event the housing-to-nozzle weld fails or the housing fails.

GE has not provided the design pressure of the automatic depressurization system accumulators for the staff to determine whether they should or should not be considered as credible missiles. Moderate energy vessels are not considered credible missiles.

GE evaluated the potential for gravitational missiles inside containment. Non safety-related components are seismically supported to prevent their collapse during a safe shutdown earthquake. These components include all cable trays for both Class 1E and non-Class 1E circuits. The components also include non-class 1E conduits, and non safety-related piping that are identified as potential hazards to safety-related equipment. Also, equipment for maintenance will either be removed during operation or seismically restrained to prevent it from becoming a missile.

Containment walls, floors and slabs also provide protection from missiles generated by the failure of rotating equipment and pressurized components. Penetration of these structures by such missiles is not considered to be credible. The adequacy of these structures for missile protection is evaluated in Section 3.5.3 of this SER. GE has not listed containment structures and slabs which provide missile protection. Credible secondary missiles (concrete fragments) due to the impact of primary missiles on containment structures are addressed in Section 3.5.4, "Interfaces", of this SER.

Based on the above, the staff concludes that the ABWR design is in conformance with GDC 4 as it relates to protection against internally generated missiles inside the containment except for lack of information on automatic depressurization system accumulators and containment structures and slabs. The design of the facility for providing protection against internally generated missiles meets the acceptance criteria of SRP Section

3.5.1.2 except as noted above.

3.5.1.4 Missiles Generated by Natural Phenomena

The staff has reviewed the protection for facility structures, systems, and components important to safety from missiles generated by natural phenomena in accordance with SRP Section 3.5.1.4. 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 provided in the "Review Procedures" portion of the SRP section. The acceptance criteria formed the basis for the staff's evaluation of the design of the facility for protecting the structures, systems, and components against missiles generated by natural phenomena with respect to the applicable regulations of 10 CFR Part 50. The SRP acceptance criteria specify that the design meet GDC 4 and GDC 2, "Design Basis for Protection Against Natural Phenomena." GDC 2 requires that structures, systems, and components important to safety be protected from the effects of natural phenomena. GDC 4 requires that these structures, systems, and components be protected against missiles. The design is considered to be in compliance with GDCs if it meets the guidelines of Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants," Positions C.1 and C.2 and Regulatory Guide 1.117, "Tornado Design Classification," Positions C.1 through C.3.

GE considered that tornado-generated missiles are the only limiting natural phenomena hazard in the design of structures, systems, and components important to safety. The staff agrees with the above position as a reasonable design basis for a standard plant. The missiles considered in the design are taken from ANSI/ANS-2.3. GE has not identified the tornado region and the list of design basis tornado-generated missiles. GE has not discussed how the design of the structures, systems, and components meets the guidelines of Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants," Positions C.1 and C.2 and Regulatory Guide 1.117, "Tornado Design Classification," Positions C.1 through C.3. Therefore, demonstration of compliance of the structures, systems, and components design with GDCs 2 and 4 remains an open item.

For individual applicants referencing the ABWR design, the staff will require them to (1) identify the missiles generated by other site-specific natural phenomena (e.g., flood) that can be more limiting than those considered in the ABWR design, and (2) provide protection for the structures, systems, and components in their facilities against such missiles.

3.5.2 Structures, Systems, and Components To Be Protected From Externally Generated Missiles

The design of the facility for protecting structures, systems, and components important to safety (within the ABWR design scope) against externally-generated missiles was reviewed in accordance with SRP Section 3.5.2. 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 provided in the "Review Procedures" portion of the SRP section. The acceptance criteria formed the basis for staff's evaluation of the design of the facility for protection against externally-generated missiles with respect to the applicable regulations of 10 CFR Part 50. The SRP acceptance criteria require the design meet GDCs 2 and 4. GDC 2 requires that structures, systems, and components important to the safety of the plant be protected from the effects of natural phenomena. GDC 4 requires that these structures, systems, and components be protected from the effects of externally generated missiles. The design is considered to be in compliance with the GDCs if it meets the guidelines of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," as related to the spent fuel pool systems and structures being capable of withstanding the effects of externally generated missiles and preventing missiles from contacting stored fuel assemblies; Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," as related to the ultimate heat sink and connecting conduits being capable of withstanding the effects of externally generated missiles; Regulatory Guide 1.115 as related to the protection of the structures, systems, and components important to safety from the effects of turbine missiles; and Regulatory Guide 1.117 as related to the protection of structures, systems, and components important to safety from the effects of tornado missiles. Protection from the low-trajectory turbine missiles, including compliance with the guidelines of Regulatory Guide 1.115 is discussed in Section 3.5.1.3 of this SER.

GE has identified safety-related structures, systems, and components in ABWR SSAR Table 3.2-1. GE has considered the tornado-generated missiles to be the limiting externally generated missiles for the ABWR design. Therefore, all the safety-related systems and components listed in the table are located in tornado resistant buildings or structures. The new and spent fuel storage systems are located in the tornado resistant reactor building.

The ultimate heat sink and the connecting conduits are not considered in this section. GE has identified these as being outside the scope of the ABWR and has imposed an interface requirement for them (see Section 3.5.4 of this SER) for the applicants referencing the ABWR standard plant.

As stated in SER Section 3.5.1.4, GE has not provided a list of the design basis tornado-generated missiles considered in the ABWR design. Also, GE has not given information on design features provided to protect (1) the charcoal delay tanks against design basis tornado missiles, and (2) all non safety-related structures, systems, and components whose failure could adversely impact the safety function of safety-related structures, systems, and components. Until the above information is provided, the staff cannot conclude that the ABWR design complies with GDCs 2 and 4 with respect to protecting the structures, systems, and components important to safety from externally generated missiles. The adequacy of barriers and structures provided in the ABWR design for protecting structures, systems, and components important to safety from externally generated missiles is evaluated in Section 3.5.3 of this SER.

3.5.4 Interfaces

ABWR SSAR Section 3.5.4, "Interfaces," lists the missile protection related interfaces which an individual applicant referencing the ABWR design must meet.

ABWR Subsection 3.5.4.1, "Protection of Ultimate Heat Sink," requires the individual applicant referencing the ABWR design to meet Positions C.2 and C.3 of Regulatory Guide 1.27 to demonstrate that the ultimate heat sink and the connecting conduits are capable of withstanding the effects of externally generated missiles.

ABWR SSAR Subsection 3.5.4.2, "Missiles Generated by Natural Phenomena from remainder of Plant Structures, Systems and Components," requires the individual applicant referencing the ABWR design to analytically check the remainder of plant structures, systems, and components (outside the scope of ABWR) to ensure that during a site specific tornado, these structures, systems, and components do not generate missiles more limiting than the tornado generated missiles considered in the ABWR design. The only credible source of secondary missiles inside containment is from the formation of concrete fragments due to impact of primary missiles with structural walls and slabs. The facility must be designed to provide protection against such secondary missiles. ABWR SSAR Subsection 3.5.4.4, "Secondary Missiles Inside Containment," has identified demonstration of protection against the above secondary missiles inside containment as an interface requirement for individual applicants referencing the ABWR design.

The staff finds the above interface requirements acceptable.

3.6 Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping

3.6.1 Plant Design for Protection Against Postulated Piping Failure In Fluid Systems Outside Containment

The design of the facility for providing protection against postulated piping failures in fluid systems outside containment but within the ABWR design scope was reviewed in accordance with SRP Section 3.6.1. 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 provided in the "Review Procedures" portion of the SRP section. The acceptance criteria formed the basis for evaluating the design of the facility for protecting against postulated piping failures 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 Bases," as it relates to accommodating the dynamic effects of postulated pipe rupture, including the effects of pipe whipping and discharging fluids. The design is considered to be in compliance with GDC 4 if it conforms to Branch Technical Position (BTP) ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment", for high and moderate- energy-fluid systems outside the containment.

Protection against postulated piping failures outside containment for structures, systems, and components not within the scope of ABWR will be evaluated individually for each application referencing the ABWR design. Additional systems or features added to the ABWR structures by referencing applications will be reviewed and evaluated on a case-by-case basis.

GE evaluated the effects of postulated pipe breaks in high-energy-fluid systems with respect to pipe whip, jet impingement, flooding, room pressurization, and environmental parameters such as temperature, pressure, humidity and radiation. GE excluded pipe break consideration and the resulting dynamic effects in the postulation of piping failures in main steam and feedwater systems. GE justified the exclusion stating that the piping in these systems meets the leak before break criteria. The staff's evaluation of the exclusion is discussed in Section 3.6.3 of this SER. GE evaluated pipe leakage crack events involving moderate-energy fluid systems for wetting from spray, flooding, and other environmental effects. GE addressed the protection methods for the systems against the effects of piping failures. Physical separation is used to the extent practicable to prevent the loss of redundant essential systems (including auxiliaries) from any single postulated event. If spatial separation requirements (based on specific breaks) between redundant trains or systems cannot be maintained, specific barriers, enclosures, shields, or restraints are provided.

Protection also includes in ensuring that the equipment and components important to safety are environmentally qualified for the environment to which they may be exposed as a result of postulated piping failures. GE provided Appendix 3I, "Equipment Qualification Environmental Design Criteria," as part of ABWR SSAR. The staff's evaluation of the protection provided against adverse environmental effects resulting from postulated piping failures is discussed in Section 3.11 of this SER.

GE has not provided a response to the staff's request for additional information dated September 12, 1988 on a number of issues, e.g., identification of all high and moderate energy piping; the impact of moderate-energy piping leakage cracks on safety-related systems and protection for them from impact; justification for the non-inclusion of process sampling system, fire protection system, HVAC emergency cooling water system, and reactor building cooling water system; specific protection features such as barriers, shields and pipe whip restraints; the results of an analysis of postulated worst case piping failure of a moderate or high-energy line for the RCIC compartment, equipment and valve room and other applicable areas outside of the containment (e.g., RHR piping areas); and, subcompartment analysis for the steam tunnel. The staff is concerned that the main steam and feedwater lines are routed via a tunnel through the control building. A steam or feedwater line break can render the control room inhabitable and compromise the safe shutdown capability. Further, the ABWR interface requirements for the referencing applicants do not call for a submittal of flooding analyses for the control room. Postulated leakage cracks in essential service water or reactor building cooling water system piping can adversely impact the control room habitability systems. Until the response to all the requested information is provided and found acceptable, and the other issues identified above are resolved, the staff cannot conclude that the ABWR design for protection against postulated piping failures outside the containment complies with GDC 4.

3.11 Environmental Qualification of Electrical Equipment Important to Safety and Safety-Related Mechanical Equipment

3.11.1 Introduction

Equipment that is used to perform a necessary safety function must be demonstrated to be capable of maintaining functional operability under all service conditions postulated to occur during its installed life for the time it is required to operate. This requirement--which is embodied in General Design Criteria (GDC) 1 and 4 of Appendix A and Sections III, XI, and XVII of Appendix B to 10 CFR 50--is applicable to equipment located inside as well as outside containment. More detailed requirements and guidance relating to the methods and procedures for demonstrating this capability for electrical equipment have been set forth in 10 CFR 50.49 "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants"; NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment" which supplements the Institute of Electrical and Electronics Engineers (IEEE) Standard 323 and various NRC Regulatory Guides (RGs) and industry standards; and Regulatory Guide 1.89 Revision 1 "Environ- mental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants."

3.11.2 Background

NUREG-0588 was issued in December 1979 to promote a more orderly and systematic implementation of equipment qualification programs by industry and to provide guidance to the NRC staff for its use in ongoing licensing reviews. The positions contained in the NUREG provide guidance on (1) how to establish environmental service conditions, (2) how to select methods that are considered appropriate for qualifying equipment in different areas of the plant, and (3) other areas such as margin, aging, and documentation.

A final rule on environmental qualification of electrical equipment important to safety for nuclear power plants became effective on February 22, 1983. This rule, section 50.49 of 10 CFR 50, specifies the requirements to be met for demonstrating the environmental qualification of electrical equipment important to safety located in a harsh environment. Regulatory Guide 1.89 Revision 1 (June 1984) identifies the guidelines that have to be met for complying with the above rule. In conformance with 10 CFR 50.49, electrical equipment for BWRs referencing the ABWR standard design may be qualified according to the criteria specified in Category I of NUREG-0588 and Regulatory Guide 1.89.

The qualification requirements for mechanical equipment are principally contained in Appendices A and B of 10 CFR 50. The qualification methods defined in NUREG-0588 can also be applied to mechanical equipment.

To document the degree to which the environmental qualification program for the ABWR complies with the NRC environmental qualification requirements and criteria, GE provided ABWR SSAR Section 3.11 "Environmental Qualification of Safety-Related Mechanical and Electrical Equipment," a response dated January 13, 1989 to the staff's request for additional information dated September 12, 1988, and ABWR SSAR Appendix 3I (proprietary) "Equipment Qualification Environmental Design Criteria."

3.11.3 Staff Evaluation

The staff evaluation of the environmental qualification program for the standard ABWR design is limited to a review of GE submittals on their approach for selection and identification of equipment required to be environmentally qualified for the ABWR, qualification methods proposed, completeness of information provided in Appendix 3I tables on environmental conditions postulated in different areas of the plant under various plant conditions, and the adequacy of the identified interface requirements. The criteria described in Section 3.11 of the NRC Standard Review Plan (NUREG-0800), Revision 2, in NUREG-0588 Category I, Regulatory Guide 1.89, Revision 1, and the requirements in 10 CFR 50.49 formed the bases for the staff's evaluation. For referencing applicants, the staff will review specific details of the environmental qualification program for their plants using the same evaluation bases mentioned above.

3.11.3.1 Completeness of Electrical Equipment Important to Safety 10 CFR 50.49 Items (b)(1), (b)(2), and (b)(3) identify three categories of electrical equipment important to safety that must be qualified in accordance with the provisions of the rule.

(b)(1) safety-related electrical equipment (equipment relied on to remain functional during and following design-basis events).

(b)(2) nonsafety-related electrical equipment whose failure under the postulated environmental conditions could prevent satisfactory accomplishment of the safety functions by the safety-related equipment.

(b)(3) certain post-accident monitoring equipment (R.G. 1.97, Category I and 2 post-accident monitoring equipment).

GE stated that for the ABWR, all three categories of electrical equipment mentioned above which are located in a harsh environment will be environ- mentally qualified. GE has identified an interface for referencing applicants which requires them to list all electrical equipment within the scope of 10 CFR 50.49 in their plant-specific environmental qualification documents (EQDs). Based on the above, the staff finds GE's approach for selection and identification of electrical equipment required to be environmentally qualified for the ABWR acceptable. The staff will review specific details provided by referencing applicants to demonstrate their compliance with 10 CFR 50.49b(1), (b)(2) and (b)(3) requirements with respect to completeness of electrical equipment important to safety required to be environmentally qualified. The details will include the list of systems and their components included in the plant environmental qualification program and the design features for preventing the potential adverse consequence identified in IE Information Notice No. 79-22 "Qualification of Control Systems."

3.11.3.2 Qualification Methods

3.11.3.2.1 Electrical Equipment in a Harsh Environment
Detailed procedures for qualifying safety-related electrical equipment in a harsh environment are defined in NUREG-0588 and Regulatory Guide 1.89. The criteria in these documents are also applicable to other equipment important to safety defined in 10 CFR 50.49.

The General Electric (GE) Environmental Qualification Program presented in GE Topical Report NEDE-24326-1-P outlines the methodology used by GE to qualify nuclear steam supply system (NSSS) safety-related electrical equipment subject to a harsh environment. GE has adopted this program for the ABWR (ABWR SSAR Subsection

3.11.2).

Based on its review of the topical report, the staff found the qualification methodology presented in the report conforms to 10 CFR 50.49 requirements and its associated standards with the exception of the position on time margin identified in the report. NUREG-0588 states that the time margin for certain categories of equipment (these categories are identified in the NUREG) should be a minimum of one hour. Neither the topical report, nor the GE submittals on the ABWR to date, have addressed the above requirement. The staff considers the time margin issue an open item and requires it to be resolved in accordance with NUREG-0588 requirements or as amplified in Regulatory Guide 1.89, Revision 1.

GE has also identified an interface for referencing applicants which requires them to identify, in their EQDs, the specific methods they will use to qualify various electrical equipment important to safety located in harsh as well as mild environments.

Based on the above, the staff finds the generic information provided by GE on environmental qualification methods for the ABWR standard design acceptable, subject to resolution of the time margin issue identified above. The staff will review the specific methods that referencing applicants will use for qualifying various electrical equipment important to safety.

3.11.3.2.2 Safety-Related Mechanical Equipment in a Harsh Environment. Although there are no detailed requirements for mechanical equipment, GDC 1, "Quality Standards and Records," and 4, "Environmental and Dynamic Effects Design Bases," and Appendix B to 10 CFR 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants" (Section III, "Design Control," and XVII, "Quality Assurance Records"), contain the following requirements related to equipment qualification:

Components shall be designed to be compatible with the postulated environmental conditions, including those associated with LOCAs.

Measures shall be established for the selection and review for suitability of application of materials, parts, and equipment that are essential to safety-related functions.

Design control measures shall be established for verifying the adequacy of design.

Equipment qualification records shall be maintained and shall include the results of tests and materials analyses.

Regarding the qualification program for safety-related mechanical equipment for the ABWR design, GE stated that the program will include all safety-related mechanical equipment identified in ABWR SSAR Section 3.2. GE further stated the following: "The mechanical equipment qualification program to be applied to the ABWR will use applicable portions of the NRC approved Licensing Topical Report NEDE-24326-1-P and Regulatory Guide 1.89, Revision 1; and will be consistent with the program for qualification of mechanical equipment in a harsh environment described in the NRC approved GESSAR II design. The ABWR program scope looks not only at the metallic components of the equipment but also the nonmetallic components. Metallic components which form a pressure boundary are considered to be qualified by the nature of their pressure retention capability as demonstrated by the application of an ASME Boiler and Pressure Vessel Stamp. Nonmetallic [components], such as greases, gaskets, lubricants, etc., will be shown to be capable of performing their intended functions under accident environments. The design of safety-related mechanical equipment associated with the ABWR will be performed under the same internal procedural controls as that used for the design of mechanical components associated with the

GESSAR II design. These controls assure that components are designed to be compatible with their postulated operating environments, that measures are established for the selection and review of the suitability of application of the material, parts, and equipment that are essential to safety-related functions, and that there are design control measures for verifying the adequacy of the design. As stated in NEDE-24326-1-P, a complete set of qualification records are developed for each safety-related component."

GE has also identified interfaces for the referencing applicants. The interfaces require the referencing applicants to provide in their plant- specific EQDs (1) a list of all safety-related mechanical equipment which are located in harsh environment plant zones, and (2) the methodology that will be used to qualify the equipment located in harsh as well as mild environment plant zones.

Based on the above, the staff finds the information provided by GE on selection and identification of mechanical equipment required to be environmentally qualified and the qualification methods for the equipment for the ABWR standard design acceptable, subject to the resolution of the time margin issue as it applies to safety-related mechanical equipment. The staff will review specific details of the environmental qualification program as it relates to safety-related mechanical equipment for referencing applicants to demonstrate their compliance with the applicable requirements for such equipment.

3.11.3.3 Completeness of Information Provided in Appendix 3I Tables

ABWR SSAR Section 3.11 defines all the environmental conditions to which the applicable equipment may be exposed during the plant operation. These are normal, abnormal, test, accident and post-accident environmental conditions. ABWR SSAR Appendix 3I contains 21 tables which specify the design limits or time-based profile of thermal environmental parameters (pressure, temperature and relative humidity), and/or design limits for radiation environmental parameters (gamma dose rate and total gamma integrated dove) for each plant area or zone in the area under normal and/or abnormal/accident environmental conditions. Additionally, the applicable tables include the neutron flux during normal operating conditions for different zones of the primary containment. The areas for which the environmental data are tabulated include the primary containment, secondary containment portion of the reactor building, remaining portions of the reactor building, turbine building, control building, radwaste building, service building and outdoor area. Except for the radwaste building and the outdoor area, all other areas have been further subdivided into zones on the basis of thermal and radiation environmental conditions determined for the zones. GE considered a postulated reactor coolant (steam or water) pressure boundary pipe rupture as the limiting accident for calculating the design limits or the time-based profile for thermal environmental parameters during accident conditions for all zones. GE considered the design basis LOCA as the limiting accident for calculating the design limits for radiation environmental parameters during accident conditions for all applicable zones. GE computed the total normal and accident doses in a zone by integrating the ambient dose rate in the zone over a 60-year period and the accident dose rate in the zone over a 6 month period, respectively.

GE stated that the environmental conditions identified in the Appendix 3I tables are upper bound envelopes for these conditions in various areas or zones to which the applicable equipment has to be designed and qualified. The environmental parameters specified in these tables are for the upper bound envelopes. GE stated that the parameters do not include margins needed to satisfy equipment qualification requirements. GE further stated that these tables include identification of significant enveloping abnormal conditions and each enveloping accident event that impacts the zone environment. GE has provided the appendix 3I tables for use by referencing applicants in developing their plant-specific environmental qualification programs for equipment important to safety.

The staff has reviewed the Appendix 3I tables and finds a number of deficiencies. Contrary to what has been stated in the ABWR Subsection 3.11.1 or Appendix 3I Subsection 3I.3.1.2, the Appendix 3I tables do not include chemical environmental conditions (chemical composition and the resulting pH) to which the applicable equipment may be exposed during normal and accident conditions, and the beta radiation dose rate and the integrated beta dose for applicable zones. The tables do not identify zone by zone whether the subject zone is environmentally mild or harsh and also do not list the typical equipment located in each zone. The environmental conditions during abnormal plant operational conditions have been put under abnormal/accident conditions in the tables. It is not clear whether in developing the environmental data for these tables for applicable zones, the adverse environmental conditions resulting from abnormal events such as SRV discharges and loss of nonsafety- related HVAC and their durations have been considered. The tables do not explicitly identify the limiting accident (e.g., high-energy line break such as main steam line, RCIC, RHR or RWCU line break; design basis LOCA inside the containment) for each zone (e.g., steam tunnel, RHR pump room) that results in the most severe environment, particularly thermal, in the zone. The tables also do not contain information on the environmental conditions due to spray or submergence and consequent wetting of equipment in applicable zones arising from piping failures and the duration of the spray or submergence. Additionally, the tables do not contain (1) radiation environmental data under normal plant operating conditions for the radwaste building, outdoor area and control building zones, (2) thermal environmental data under accident conditions for the radwaste building, service building zones, and outdoor area, and (3) radiation environmental data under accident conditions for the turbine building zones, radwaste building, service building zones, and outdoor area.

If some of the areas identified above are not expected to house any equipment required to be qualified and therefore environmental data need not be provided for them, it should be so stated. The tables do not contain sufficient information on thermal environmental conditions (e.g., duration of different conditions) in various zones under normal plant operational conditions to develop a meaningful time-based thermal environmental profile for the zones. There is no consistency in the units employed to specify the pressures (e.g., kg/cm g; mm Aq). It is not clear what the statement "the pressure will be kept negative or positive" means (see Note 2 to Tables 3I.3-3 through 3I.3-7). It is not clear whether in developing the gamma dose limits in applicable areas (e.g., secondary containment zones), the potential for exposure of equipment in the areas due to recirculating fluid lines has been considered. The staff notes that the integrated gamma accident dose in the primary containment for the ABWR compares with the typical value of about 2×10^8 rads quoted in the safety analysis reports of several operating reactors (e.g., Perry: 2.7×10^8 rads; River Bend: 1.7×10^8 rads; Clinton: 2×10^8 rads; Nine Mile Point: 1.4×10^8 rads). It is not clear why the ABWR integrated gamma accident dose is lower than the corresponding doses quoted for several operating reactors.

Based on its review of the Appendix 3I tables discussed above, the staff concludes that though the Appendix 3I tables contain some useful information, they are incomplete and will, therefore, provide only marginal guidance to the referencing applicants in developing a sound environmental qualification program for applicable equipment in their plants. The staff requires GE to address all the above concerns and revise the tables as appropriate. Until then, the staff cannot conclude that the Appendix 3I tables contain adequate information.

3.11.3.4 Adequacy of Interface Requirements

Besides the interface requirements identified in Subsections 3.11.3.1, 3.11.3.2.1 and 3.11.3.2.2 of this DSER, GE has identified two additional interfaces for the applicants referencing the ABWR design. In these instances applicants shall (1) present a summary of environmental conditions and qualified conditions for each applicable item of equipment located in a harsh environmental zone in the system component evaluation work (SCEW) sheets as described in Table I-1 of GE Topical Report

NEDE-24326-1-P and compile these sheets in their plant-specific EQDs, and (2) record and maintain in an auditable file, the results of all qualification tests for applicable equipment. Additionally, though not identified as an interface requirement, GE states that a surveillance and maintenance program will be developed for each applicable equipment item located in a mild environment zone to ensure its operability during its designed life. GE has also identified a requirement for the vendors of mild environment equipment. It requires the vendors to submit a certificate of compliance certifying that the subject equipment has been qualified for the requirements identified to assure its capability to perform its safety-related function in its applicable environment.

Based on the above, the staff finds the identified interface requirements for referencing applicants and other requirements mentioned above acceptable.

3.11.3.5 Conclusion

Based on its review of all the GE submittals to date (see Subsection 3.11.2 of this DSER), the staff concludes that the proposed environmental qualification program for ABWR electrical equipment important to safety and safety-related mechanical equipment is acceptable, subject to resolution of all the concerns identified in Subsections 3.11.3.2.1, 3.11.3.2.2 and 3.11.3.3 of this SER.

For individual plants referencing the ABWR design, the staff will evaluate the specific details of their plant-specific environmental qualification program, including maintenance and surveillance for applicable equipment located in harsh environment zones.

PLANT SYSTEMS BRANCH
SECTIONS 6.2, 6.2.1, 6.2.2, 6.2.3, 6.2.4, AND 6.2.5

Enclosed is input to the draft SER for the subject ABWR SSAR sections identified above. Most of the subject sections have quite a few open items that are noted in the enclosed SER. These open items (as listed below) will require additional information:

1. With respect to the analytical models used for containment pressure and temperature analysis, GE has not provided a detailed discussion to describe how the two ABWR drywell volumes and the combination vertical and horizontal vent system are modeled in the computer code to represent the physical geometry of the containment. In addition, GE has not provided tests performed on the combination vertical and horizontal drywell-to-wetwell vent system to demonstrate air carry-over, vent clearing, condensation and chugging predicted in the suppression pool following a LOCA (SER Section 6.2.1.2.1).

2. Pending acceptability of the analytical model, we have not been able to conclude that the containment pressure and temperature response analysis is acceptable (SER Section 6.2.1.3).

3. With regard to drywell depressurization and the provision of a wetwell-to- drywell vacuum relief system, GE has not provided detailed information about the analytical model (including condensation heat transfer, modeling the drywell spray and ECCS spill, etc.) for our review. Furthermore, GE has not identified the specific vacuum breakers and their arrangement details and has not proposed a test program to demonstrate that they will perform as predicted (SER Section 6.2.1.5.1).

4. With regard to the suppression pool dynamic loads, GE has recently provided Appendix 3B "Containment Loads," for our review. We are currently reviewing Appendix 3B and will provide our evaluation on completion of the review (SER Section 6.2.1.6).

5. During the course of its review, the staff requested GE to provide more detailed information with regard to the subcompartment pressure analysis and steam bypass of the suppression pool. To date, the staff has not received such information (SER Sections 6.2.1.7 and 6.2.1.8).

6. With regard to the secondary containment functional design, GE has not discussed how the guidelines of SRP 6.2.3, "Secondary Containment Functional Design," will be met (SER Section 6.2.3).

7. With regard to potential secondary containment bypass leakage, GE has been requested to provide additional information to justify the bypass leakage paths barriers that are relied upon to preclude bypass flow. To date, the staff has not received such information (SER Section 6.2.3.1).

8. During the course of its review, the staff requested that GE provide more detailed information with respect to the containment isolation system. To date, GE has provided the response, in part, to justify deviations for some containment penetrations. As a result of its review, however, the staff finds the information provided by GE to be insufficient. The staff will discuss the information needed to complete its review with GE (SER Section 6.2.4).

9. With regard to the containment purge system, GE has not provided information on isolation valve closure times and purge/exhaust line sizes that are necessary to evaluate compliance with Branch Technical Position CSB 6-4. Also, a radiological consequence analysis for a LOCA with the purge system initially open has not been provided in accordance with CSB 6-4. In addition, GE has not addressed the structural integrity of the purge/exhaust system when subjected to LOCA thermal-hydraulic conditions. Therefore, the staff has not been able to complete its review of the ABWR containment purge system (SER Section 6.2.4.1).

10. With regard to combustible gas control in containment, GE has not provided information with respect to the ABWR combustible gas control system as requested by the staff in its RAI dated July 7, 1988 (SER Section 6.2.5).

11. GE ABWR has not explicitly addressed severe accident considerations (SER Section 6.2.6).

6.2 Containment Systems

The containment systems for the ABWR include a containment structure as the primary containment, a secondary containment (reactor building) surrounding the primary containment and housing equipment essential to safe shutdown of the reactor and fuel storage facilities, and supporting systems. The primary containment is designed to prevent the uncontrolled release of radioactivity to the environment with a leakage rate of 0.5 percent by weight per day at the calculated peak containment pressure related to the DBA. The secondary containment is designed to confine the leakage of airborne radioactive materials from the primary containment. Figure 6.2.1 shows the principal features of the ABWR containment.

6.2.1 Primary Containment Functional Design

The ABWR primary containment design has the following main features:

(1) A drywell comprised of two volumes: (a) an upper drywell (UD) volume surrounding the reactor pressure vessel (RPV) and housing the steam and feedwater lines and other connections of the reactor primary coolant system, safety/relief valves and the drywell HVAC coolers, and (b) a lower drywell (LD) volume housing the reactor internal pumps, control rod drives and under vessel components and servicing equipment. The upper drywell is a cylindrical steel-lined reinforced concrete structure with a removable steel head and a reinforced concrete steel diaphragm floor. The cylindrical RPV pedestal, which is connected rigidly to the steel diaphragm floor, separates the lower drywell from the wetwell. Ten UD to LD connecting vents (DCVs), approximately 1M X 2M in cross-section, are built into the RPV pedestal. The DCVs are extended downward via 1.2M inside diameter steel pipes, each of which has three horizontal vent outlets into the suppression pool.

The drywell, which has a net free volume of 259,563 ft. , is designed to withstand design pressure and temperature transients following a LOCA and also the rapid reversal in pressure when the steam in the drywell is condensed by emergency core cooling system flow during post LOCA flooding of the RPV. A wetwell-to-drywell vacuum relief system is provided to prevent back-flooding of the suppression pool water into the lower drywell and to protect the integrity of the steel diaphragm floor slab between the drywell and wetwell and the drywell structure and liner. The drywell design pressure and temperature are 45 psig and 340 F, respectively. The design drywell-to-wetwell differential pressures are (+) 25 psid and (-) 2 psid. The design drywell-to- reactor building negative differential pressure is (-) 2 psid.

(2) A system of drywell-to-wetwell vents which channel blowdown from the drywell and discharge into the suppression pool following a LOCA. There are 30 vents in the vertical section of the lower drywell below the suppression pool water level, each with a nominal diameter of 2.3 feet. These vents are arranged in 10 circumferential columns, each containing three vents. The three vent centerlines in each column are located at 11.48 feet, 15.98 feet and 20.48 feet below the suppression pool water level when the suppression pool is at the low water level.

(3) A wetwell, comprised of an air volume and suppression pool, with a net free air volume of 210,475 feet and a minimum pool volume of 126,427 feet at low water level. The wetwell is designed for an internal pressure of 45 psig and a temperature of 219 F. The design wetwell-to-reactor building negative differential pressure is (-) 2 psid. The suppression pool, located inside the wetwell annular region between the cylindrical RPV pedestal wall and the outer wall of the wetwell, is a large body of water which serves as a heat sink for postulated transients and accidents and as a source of cooling water for the emergency core cooling system (ECCS). In the case of transients that result in a loss of the ultimate heat sink, energy would be transferred to the pool by the discharge piping from the reactor system's safety/relief valves (SRVs). In the event of a LOCA within the drywell, the drywell atmosphere is vented to the suppression pool through the system of drywell-to-wetwell vents.

This primary containment design basically uses combined features of the Mark II and Mark III designs, with the exception that the drywell is composed of upper and lower drywell volumes. The vents to the suppression pool are a combination of the vertical Mark II and horizontal Mark III systems. The wetwell is similar to a Mark II wetwell. Table 6.2.1 provides a comparison of the design parameters for Mark I, II, III, and ABWR containments.

6.2.1.1 LOCA Chronology

Following a postulated LOCA, the drywell pressure increases as a result of blowdown of the reactor coolant system. Pressurization of the drywell causes the water initially in the vent system to be accelerated into the pool until the vents are cleared of water. During this clearing process, the water leaving the horizontal vents forms jets in the suppression pool and causes water jet impingement loads on the structures within the suppression pool and on the containment wall opposite the vents.

During the vent clearing transient, the drywell is also subjected to a pressure differential, and the RPV pedestal wall experiences a vent-clearing reaction force.

Immediately following vent clearing, an air and steam bubble forms at the exit of the vent. The bubble pressure initially is assumed equal to the existing drywell and wetwell differential pressure. This bubble transmits a pressure wave through the suppression pool water and results in a loading on the suppression pool boundaries and on equipment located in the suppression pool.

As the air and steam flow from the drywell becomes established in the vent system, the initial vent exit bubble expands to equalize the suppression pool hydrostatic pressure. GE's large-scale Pressure Suppression Test Facility (PSTF) tests show that the steam fraction of the flow is condensed, but continued injection of drywell air and expansion of the air bubble result in a rise of the suppression pool surface. During the early stages of this process, the pool swells in a bulk mode (i.e., a slug of solid water is accelerated upward by the air pressure). Structures close to the pool surface will experience loads as the rising pool surface impacts the lower surface of the structure. In addition to these initial impact loads, these same structures will experience drag loads as water flows past them. Equipment in the suppression pool will also experience drag loads.

Data from PSTF air tests indicate that after the pool surface has risen approximately 1.6 times the initial submergence of the top vent (which translates to 12 feet above the initial pool surface for the Mark III design the thickness of the water ligament could be as small as 2 feet or less, and the impact loads would them be significantly reduced. This phase is referred to as "incipient breakthrough," i.e., the ligament begins to break up. To account for possible non conservatism in the test facility arrangement and instrumentation error bands, the staff has determined that the breakthrough height should be set at 18 feet above the initial pool surface for the Mark III design. The staff's evaluation of the breakthrough height for ABWR design is discussed in Section 6.2.1.6 of this SER.

As the drywell air flow through the horizontal vent system decreases and the air/water suppression pool mixture experiences gravity-induced phase separation, pool upward movement stops and the fallback process starts. During this process, floors and other flat structures experience downward loading, and the containment wall theoretically can be subjected to a small pressure increase. However, this pressure increase has not been observed experimentally.

As the reactor blowdown proceeds, the primary system is depleted of high-energy fluid inventory and the steam flow rate to the vent system decreases. This reduced steam flow rate leads to a reduction in the drywell-to-wetwell pressure differential that, in turn, results in a sequential recovering of the horizontal vents. Suppression pool recovery of a particular vent row occurs when the vent stagnation differential pressure corresponds to the suppression pool hydrostatic pressure at that row of vents.

Toward the end of the reactor blowdown, the top row of vents is capable of condensing the reduced blowdown flow and the two lower rows will be totally recovered. As the blowdown steam flow decreases to very low values, the water in the top row of vents starts to oscillate, causing what has become known as vent chugging. This action results in dynamic loads on the top vents and on the RPV pedestal wall opposite the upper row of vents. In addition, an oscillatory pressure loading condition can occur on the drywell and wetwell. Because this phenomenon is dependent on a low steam mass flux (the chugging threshold appears to be in the range of 10 lb/sec/ft.), it is expected to occur for all break sizes. For smaller breaks, it may be the only mode of condensation that the vent system will experience.

The staff's evaluation of this LOCA-related pool dynamic loads is discussed in Section 6.2.1.6 of this SER.

Shortly after onset of a DBA, the ECCS pumps automatically start and pump suppression pool water into the reactor pressure vessel. This water floods the reactor core and, if the operator fails to follow the emergency procedure guidelines requiring ECCS flow to be throttled, the water starts to cascade into the drywell from the break. When this occurs depends on the size and location of the break. Because the drywell is full of steam at the time of vessel flooding, the sudden introduction of cool water causes rapid steam condensation and drywell depressurization. When the drywell pressure falls below the wetwell airspace pressure, air from the wetwell redistributes between the drywell and wetwell via the wetwell-to-drywell vacuum relief system. Eventually enough air returns to equalize the drywell and wetwell pressures, however, during this drywell depressurization transient, there is a period of negative pressure on the drywell structure. A negative load condition of (-) 2 psid is, therefore, specified for drywell design.

The staff's evaluation of this drywell to wetwell negative differential pressure is discussed in Section 6.2.1.5.1 of the SER.

6.2.1.2 Containment Analysis

The staff's review of the containment design included the temperature and pressure responses of the drywell and wetwell to a spectrum of LOCAs, the capability to withstand the effects of steam bypass from the drywell directly to the air region of the suppression pool, the external pressure capability of the drywell and wetwell and the negative drywell-to-wetwell differential pressure. In addition, the review considered GE's proposed design bases and criteria for the containment, the analyses and test data in support of the criteria and bases, and the loads resulting from pool dynamic phenomena.

6.2.1.2.1 Containment Analytical Model

GE's calculation of the short term and long term containment pressure- temperature response to postulated high energy line breaks used the same analytical models and conservative assumptions that were previously presented and reviewed for the Mark III containment in GESSAR II. The staff found these to be acceptable using independent confirmatory analyses with the CONTEMP-LT28 computer code. These models and assumptions are discussed in the ABWR SSAR and NEDO-20533 and its supplement 1, "The GE. Mark III Pressure Suppression Containment Analytical Model." In a response to the staff's request for additional information (RAI), GE stated that the analytical models described in NEDO-20533 are appropriate to calculate the ABWR containment responses to postulated accidents. Though originally written for prediction of Mark III transients, these models, which simulate from first principles the transient conditions in the containment, can be adapted for the ABWR containment configuration. These models simulate the drywell, vent systems, and wetwell (suppression pool and airspace). They are, therefore, adaptable to other containment configurations having the same basic components.

As indicated in Section 6.2.1 of this SER, the ABWR containment design basically uses combined features of Mark II and Mark III designs, with the exception of a unique feature of two drywell volumes (upper and lower). The vent system is a combination of vertical (Mark II design) and horizontal (Mark III design) drywell-to-wetwell vent systems. The wetwell (suppression pool and airspace) is similar to a Mark II. However, GE has not provided a detailed discussion to describe how the two ABWR drywell volumes and the combination vertical and horizontal vent system are modeled in the computer code to represent the physical geometry of the containment, and how the air carry over from the two drywell volumes to the wetwell is treated in the computer code. The staff will need this information for its review of the ABWR containment analysis. In addition, the staff will require tests to verify that:

a) Following a LOCA, the combination of vertical and horizontal drywell-to-wetwell vent system will perform (to demonstrate venting clearing, condensation and chugging) as predicted.

b) Following a LOCA, the containment will perform (air carry-over, and containment pressure and temperature responses) as predicted by the analytical model. Based on its review, the staff has not been able to conclude that the assumptions and analytical models used to predict the containment pressure and temperature transients following a LOCA in the ABWR containment are acceptable. The staff will report the resolution of this matter in a supplement to this SER.

6.2.1.3 Short-Term Pressure Response

The maximum drywell-to-wetwell differential pressure occurs during the blowdown phase (short-term) of a LOCA. GE has performed analyses of various postulated primary system breaks, including a double-ended rupture of the main feedwater line, a double-ended rupture of the main steam line, and small break accidents. Results of the analyses indicate that the main feedwater line break (FWLB) yields the limiting drywell-to-wetwell differential pressure and peak drywell and wetwell pressure and is, therefore, the design-basis accident for the drywell and wetwell. The main steam line break (MSLB) yields the limiting drywell temperature. GE has provided comparative plots of drywell and wetwell short term pressure and temperature response to design basis, 0.5 ft. , 0.1 ft. , and 0.01 ft. breaks in both the main feedwater and main steam line piping inside the drywell. These figures substantiate the large guillotine breaks resulting in the highest drywell and wetwell pressure and temperature. However, these figures comparing different size pipe breaks do not indicate the same value of peak drywell and wetwell pressure as reported in Table 6.2-1 of the ABWR SSAR. The staff will require these discrepancies to be clarified. Table 6.2.2 of this SER shows the maximum calculated and design pressure and temperature in drywell and wetwell.

Standard Review Plan (SRP) Section 6.2.1.1.c, "Pressure-Suppression Type BWR Containments," states that for Mark III plants at the construction permit stage, the containment design pressure should provide at least a 15 percent margin above the peak calculated containment pressure, and the design differential pressure between drywell and containment should provide at least a 30 percent margin above the peak calculated differential pressure. GE's calculated drywell peak pressure for the FWLB is 39 psig and maximum calculated temperature is 338 F resulting from the MSLB. The design pressure for the drywell is 45 psig which provides a margin of 15 percent above the peak calculated pressure in the drywell and is equal to the margin recommended in the SRP. Therefore, the staff finds this design margin for containment pressure acceptable.

The calculated wetwell peak pressure and maximum temperature are 26 psig and 207 F (which is 12 F below the design temperature of 219 F) resulting from the FWLB. The design pressure for the wetwell is 45 psig which provides a margin of 42 percent above the peak calculated pressure in the wetwell.

The calculated drywell-to-wetwell peak differential pressure is 16 psid and the design drywell-to-wetwell differential pressure is 25 psid which provides a design margin of 56 percent. Based on its review and pending the acceptability of GE's analytical models as described in Section 6.2.1.2.1, the staff concludes that the containment pressure and temperature transients following a LOCA in the ABWR containment are acceptable. The staff will report the resolution of this matter in a supplement to this SER.

6.2.1.4 Long-Term Response

Following the short-term blowdown phase of the accident, the suppression pool temperature and containment pressure continuously increase because of the input of decay heat and sensible energy into the containment. During this period, the emergency core cooling system (ECCS) pumps, which take suction from the suppression pool, reflood the reactor pressure vessel up to the level of the main steam nozzles. Subsequently, ECCS water flows out of the break and fills the drywell establishing a recirculation flow path for the ECCS. The relatively cold ECCS water condenses the steam in the drywell and brings the drywell pressure down rapidly. After approximately 10 minutes, the residual heat removal (RHR) heat exchangers are automatically activated to remove energy from the containment via recirculation cooling of the suppression pool with the RHR service water system. This is a conservative assumption since the RHR design permits automatic initiation of containment cooling well before a 10 minute period. The containment spray is also conservatively assumed not to be used.

In the long-term analysis, GE accounted for potential post-accident energy sources. These included decay heat, pump heat rate, sensible heat, and metal-water reaction energy. GE's long-term model also assumed that the containment atmosphere would be saturated and equal to the suppression pool temperature at any time. Therefore, the containment pressure is equal to the sum of the partial pressure of air and the saturation pressure of water corresponding to the pool temperature.

Based on the above assumptions, GE calculated a peak suppression pool temperature of 206.46 F. The calculated long-term secondary peak containment drywell and wetwell pressures are well below the calculated short term peak pressures. Based on its review, and pending the acceptability of GE's analytical models as described in Section 6.2.1.2.1 of this SER, the staff finds GE's analysis for long-term response following a LOCA in the ABWR containment acceptable.

6.2.1.5 Reverse Containment Pressurization

Certain events in the primary containment cause depressurization transients that can create negative drywell-to-wetwell, drywell-to-reactor building, or wetwell-to-reactor building pressure differentials. Therefore, vacuum relief provisions may be necessary in order to limit these negative pressure differentials within design values. The events which cause containment depressurization are:

(1) Inadvertent drywell/wetwell spray actuation during normal operation,

(2) Post-LOCA drywell depressurization as a result of condensation of the steam by the spilled ECCS subcooled water, and

(3) Wetwell spray actuation following a stuck open relief valve.

6.2.1.5.1 Drywell Depressurization

Drywell depressurization, which will create a negative drywell-to-wetwell pressure differential and/or a negative drywell-to-reactor building pressure differential, is caused by two major events:

(1) Post-LOCA drywell depressurization as a result of condensation of the steam by the spilled ECCS subcooled water, and

(2) Inadvertent drywell spray actuation during normal operation.

GE indicates that drywell depressurization following a feedwater line break results in the severest negative pressure transient in the drywell. Without the provision of vacuum relief, this negative pressure transient may create a drywell-to-wetwell negative pressure differential of (-) 40 psid. This pressure differential is much greater than the design negative drywell-to-wetwell pressure difference of (-) 2 psid. Therefore, this transient is used to determine the size and the number of wetwell-to-drywell vacuum breakers.

Based on its analysis, GE further indicates that with a typical vacuum breaker diameter of 20 inches, a loss coefficient, K, of 3, and one single failure, eight wetwell-to-drywell vacuum breakers are required to maintain the negative pressure differentials of drywell-to-wetwell and of drywell-to- reactor building below the design negative pressure differentials of (-) 2 psid. However, GE has not provided detailed information with regard to the analytical model (including condensation heat transfer, modeling the drywell spray and ECCS spill, etc.) for staff review. Furthermore, GE has not identified the specific vacuum breakers and their arrangement details (e.g., <D or UD, 2 valves in series for bypass single failure protection) and has not proposed a test program to demonstrate that they will perform as predicted. Therefore, the staff has not been able to conclude that the number and arrangement of wetwell-to-drywell vacuum breakers provided for the ABWR are acceptable. The staff will report the resolution of this matter in a supplement to this SER.

6.2.1.5.2 Wetwell Depressurization

Wetwell depressurization, which will create a negative wetwell-to-reactor building negative pressure differential, is caused by the following events:

(1) Drywell and wetwell spray actuation during normal operation,

(2) Wetwell spray actuation subsequent to stuck open relief valve, and

(3) Drywell and wetwell spray actuation following a LOCA.

GE indicates that the limiting negative pressure transient in the wetwell corresponds to wetwell spray actuation following a stuck open relief valve. The effect of relief valve discharge on the suppression pool is to heat the wetwell airspace, thus increasing its pressure. When the pressure in the wetwell becomes greater than the drywell pressure, the wetwell-to- drywell vacuum relief system allows the flow of air from the wetwell to the drywell, thereby pressurizing both drywell volumes. Wetwell pressure and temperature peak when the reactor decay heat decreases below the heat removal capability from continued pool cooling and wetwell spray. Wetwell temperature and pressure decrease, but the drywell pressure remains at its peak value. When the pressure difference between the two volumes becomes greater than the hydrostatic head of water above the top vent, air flows back into the wetwell airspace, slowing down wetwell depressurization. The pressure differential between the drywell and the wetwell is maintained constant at the hydrostatic head above the top row of horizontal vents. The final pressure in the wetwell is lower than the drywell pressure because more air is transferred to the drywell during wetwell pressurization than is received during wetwell depressurization.

Inadvertent drywell or wetwell spray actuation during normal operation can cause depressurization of the sprayed volume due to the resultant condensation of vapor present in the air space. However, the magnitude of this depressurization is less than the post-LOCA or stuck-open relief valve cases because of the relatively smaller mass of condensable gas present during normal operation.

Calculation of the peak wetwell-to-reactor building negative differential pressure is based on an energy balance of the containment atmosphere before and after spray activation, assuming that the final air-vapor mixture is at 100 percent relative humidity and that there are no reactor building-to-wetwell vacuum breakers. Using these assumptions, the peak calculated wetwell-to-reactor building negative differential pressure was determined to be -1.77 psid. This is 10 percent less than the design value of -2.0 psid. The staff has reviewed the initial conditions and assumptions used in the analysis and finds them acceptable.

6.2.1.6 Suppression Pool Dynamic Loads

In a response to the staff's RAI with regard to suppression pool dynamic loads following a LOCA, GE indicated that the pool dynamic loads (such as vent clearing, pool swell, condensation oscillation, and chugging) resulting from postulated loss-of-coolant accidents and safety relief valve actuation during transients, are to be found in Appendix 3B, "Containment Loads," to the ABWR Standard Safety Analysis Report. GE has recently provided Appendix 3B for staff review. The staff is currently reviewing Appendix 3B with respect to suppression pool dynamic loads, and will report its finding in a supplement to this SER on completion of the review.

6.2.1.7 Subcompartment Pressure Analysis

Internal structures within both the drywell and wetwell form subcompartments or restricted volumes that are subjected to differential pressure subsequent to postulated pipe ruptures.

In the drywell there are two such volumes: (1) the reactor pressure vessel annulus, which is the annular region formed by the reactor pressure vessel and the biological shield, and (2) the drywell head, which is a cavity surrounding the reactor pressure vessel head. There is also a main steam tunnel located in the drywell.

The design of the containment subcompartments was based on the postulated worst-case design-basis accident (DBA) occurring in each subcompartment. For each containment subcompartment in which high-energy lines are routed, mass and energy release data corresponding to a postulated line break were calculated. All breaks were considered to be full double-ended circumferential breaks.

During the course of its review, the staff requested GE to provide more detailed information with regard to the subcompartment pressure analysis. Until the staff has received and reviewed such information, this will be an open issue and will be discussed in an SER supplement.

6.2.1.8 Steam Bypass of the Suppression Pool

The concept of the ABWR pressure-suppression containment is that steam released from the primary system will be condensed by the suppression pool and will therefore limit pressurization of the containment system. This is accomplished by channeling the steam into the suppression pool through a vent system. Bypass leakage paths could exist between the drywell and the wetwell airspace that might overpressurize the containment. Potential sources of steam bypass include cracking of the drywell concrete structure and penetrations through the drywell structure. To mitigate the consequences of any steam which may bypass the suppression pool, the wetwell spray system will be activated. The flow rate of the wetwell spray system is 500 gpm.

The allowable bypass leakage is defined as the amount of steam which could bypass the suppression pool without exceeding the wetwell design pressure. The allowable value has been evaluated by GE for the complete spectrum of credible primary system pipe ruptures. It is expressed in terms of the parameter (A/K) where:

A = flow area of leakage path (ft.)
K = geometric and friction loss coefficient

This parameter (A/ K) is dependent only on the geometry of the drywell leakage paths and is a convenient numerical definition of the overall drywell leakage capability.

GE evaluated the bypass capability of the primary containment for small primary system breaks considering containment sprays and containment heat sinks as means of mitigating the effects of bypass leakage. This analysis results in an allowable drywell leakage capability, A/ K, of 0.05 ft. , which is identical to that for the Mark II design. GE has not provided the spectrum of breaks analyzed and the limiting break size that determined the drywell leakage capacity. In addition, GE has not identified all lines from which leakage (or rupture) could contribute pool bypass and wetwell air space pressurization. Therefore, the staff has not been able to conclude that an allowable drywell leakage capability of 0.05 ft. for the ABWR is acceptable. Furthermore, since the BWR pressure suppression design is sensitive to relatively small bypass leakage areas the staff believes that advanced BWR's which use the same design concept should demonstrate a capability to tolerate bypass leakage to an amount at least equivalent to that justified for the last standardized design, i.e., the Mark III containment system. In a response to the staff's RAI, GE stated that it would provide additional information regarding pool bypass capability by December 31, 1988. To date, the staff has not received this information. Until the staff has received and reviewed such information, this will be an open issue and will be discussed in an SER supplement.

6.2.2 Containment Heat Removal System

The containment heat removal system is an integral part of the residual heat removal (RHR) system which consists of three redundant loops. Each loop is designed so that a failure in one loop cannot cause a failure in another. In addition, each of the loops and associated equipment is located in a separate protected area of the reactor building to minimize the potential for single failure, including loss of onsite or offsite power causing the loss of function of the entire system. The system equipment, piping, and support structures are designed to seismic Category I criteria.

The containment heat removal system encompasses several of the RHR operating modes, which are the low pressure flooder (LPFL) mode, the suppression pool cooling mode, and the containment (drywell and wetwell) spray modes.

a) LPFL Mode

Following a LOCA, containment cooling starts as soon as the LPFL injection flow begins. During this mode, water from the suppression pool is pumped through the RHR heat exchangers and injected into the reactor vessel. The LPFL mode is initiated automatically by a low water level in the reactor vessel or high pressure in the drywell. In addition, each loop in the RHR system can also be placed in operation by means of a manual initiation push button switch.

b) Suppression Pool Cooling Mode

Following a LOCA, the suppression pool cooling subsystem provides a means to remove heat released into the suppression pool. During this mode of operation, water is pumped from the suppression pool through the RHR heat exchangers and back to the suppression pool. This mode is initiated, as needed, manually, by closing the LPFL injection valves and opening the suppression pool return valves.

c) Containment (Wetwell and Drywell) Spray Cooling Mode

Two of the RHR loops provide containment spray cooling subsystems. Each subsystem provides both wetwell and drywell spray cooling. This subsystem provides steam condensation and primary containment atmospheric cooling following a LOCA by pumping water from the suppression pool, through the RHR heat exchangers and into the wetwell and/or drywell spray spargers in the primary containment. The drywell spray mode is initiated by operator action as needed following a LOCA by closing the LPFL injection valves and opening the spray valves.

Provisions have been made in the RHR system to permit inservice inspection of system components and functional testing of active components.

The location of suction and return lines in the suppression pool facilitates mixing of the return water with the total pool inventory before the return water becomes available to the suction lines.

Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Sumps," prohibits design reliance on pressure or temperature transients expected during a LOCA for ensuring net positive suction head. The ABWR net positive suction head design assumes 0-psig containment pressure and the maximum expected fluid temperatures resulting from a LOCA and, therefore, is acceptable.

The suppression pool make-up system provides additional water from the condensate storage tank through the suppression pool cleanup system to the suppression pool by gravity flow during normal conditions. Following a LOCA, the ECCSs take suction from the suppression pool. The quantity of water is sufficient to account for all conceivable post-accident entrapment volumes (i.e., places where water can be stored while maintaining long-term drywell vent water coverage).

Based on its review of the information in the ABWR SSAR and the responses to the staff's requests for additional information concerning the containment heat removal systems, the staff concludes that the containment heat removal systems satisfy the guidelines described in SRP Section 6.2.2, "Containment Heat Removal Systems" and Regulatory Guide 1.1, and are, therefore, acceptable.

6.2.3 Secondary Containment Functional Design

The ABWR secondary containment region completely surrounds the primary containment and is designed to remove fission products released from the primary containment during a DBA to limit whole body and thyroid doses within the guidelines of 10 CFR Part 100 and 10 CFR Part 50 Appendix A General Design Criterion 19. The two systems that fulfill this function are the secondary containment heating, ventilating, and air conditioning (HVAC) and the standby gas treatment system (SGTS). The HVAC maintains a negative pressure within the secondary containment during normal operation to prevent any radioactivity from escaping to the environment. The SGTS provides post- accident filtration and removal of airborne halogens and particulates from the secondary containment.

The components of the secondary containment are designed to withstand missiles, pipe whip, post accident environments, seismic events, a single active failure, and a loss of offsite power. The SGTS will maintain at least -0.25 inches water gage negative pressure (secondary containment to environment) after any postulated accident.

GE indicates that testing and inspection of the integrity of secondary containment will be part of the testing of the SGTS. The staff's evaluation of the SGTS is discussed in Section 6.5.3 of the SER.

SRP Section 6.2.3, "Secondary Containment Functional Design," in part, indicates that all openings, such as personnel doors and equipment hatches, should be under administrative control. These openings should be provided with position indicators and alarms having readout and alarm capability in the main control room. The effect of open doors or hatches on the functional capability of the depressurization and filtration systems should be evaluated. However, GE has not discussed how these guidelines will be met for the ABWR design.

Based on its review, the staff has not been able to find that the secondary containment functional design for the ABWR is acceptable. The staff will report the resolution of this matter in a supplement to this SER.

6.2.3.1 Secondary Containment Bypass Leakage

Although the primary containment is enclosed by the secondary containment, there are systems that penetrate both the primary and secondary containment boundaries, creating potential paths through which radioactivity in the primary containment could bypass the leakage collection and filtration systems associated with the secondary containment. A number of these lines contain physical barriers or design provisions that can effectively eliminate leakage. These include water seals, containment isolation provisions, and vent return lines to controlled regions. The criteria by which potential bypass leakage paths are determined has been set forth in Branch Technical Position CSB 6-3, "Determination of Bypass Leakage Paths in Dual Containment Plants." GE has been requested to provide additional information to justify the bypass leakage path barriers that are relied upon to preclude bypass flow. The staff will report on the resolution of this issue in a supplement to this SER.

6.2.4 Containment Isolation System

The containment isolation system includes containment isolation valves and associated piping and penetrations necessary to isolate the primary containment in the event of a LOCA. The staff's review of this system includes the number and location of isolation valves, valve actuation signals and valve control features, positions of the valves under various plant conditions, protection afforded isolation valves from missiles and pipe whip, and environmental design conditions specified in the design of components.

The staff requires that (1) the containment isolation system be designed to automatically isolate the containment atmosphere from the outside environment under accident conditions; (2) double-barrier protection be provided to ensure that no single active failure will impair the containment isolation; and (3) all components be protected from missiles, water jets, and pipe whip. The design of containment isolation provisions should conform to the requirements of GDCs 54, 55, 56, and 57, as appropriate. Justification should be provided if deviations from these requirements exist.

The piping systems of the ABWR that penetrate containment can be classified into three areas:

(1) Piping lines that meet the explicit requirements of GDCs 54, 55, 56, and 57,

(2) Piping lines that do not meet the explicit requirements of GDCs 54, 55, 56, and 57 but are acceptable based on their meeting the specific guidelines given in SRP 6.2.4, which constitute acceptable alternative design provisions, and

(3) Other lines that must be reviewed on a case-by-case basis to determine if an acceptable alternative basis exists for allowing a deviation from the explicit GDC on grounds not previously articulated in the SRP.

During the course of its review, the staff requested GE to provide more detailed information with respect to the containment isolation system. To date, GE has provided the response, in part, to justify the deviations of some of the containment penetrations from the above cited requirements. As a result of it review, however, the staff finds the information provided by GE to be insufficient.

Based on its review, the staff has not been able to find that the containment isolation system for the ABWR design is acceptable. The staff will report the resolution of this matter in a supplement to this SER.

6.2.4.1 Containment Purge System

In a response to the staff's request, GE described the general arrangement of the containment purge system. However, GE has not provided information on isolation valve closure times and the purge/exhaust line sizes that are necessary to evaluate compliance with Branch Technical Position CSB 6-4. Also, a radiological consequence analysis for a LOCA with the purge system initially open has not been provided in accordance with CSB 6-4. In addition, GE has not addressed the structural integrity of the purge/exhaust system when subjected to LOCA thermal-hydraulic conditions. Therefore, the staff has not been able to complete its review of the ABWR containment purge system. The staff will continue its evaluation of the information (when available) and will report the resolution in a supplement to this SER.

6.2.5 Combustible Gas Control in Containment

Following a LOCA, hydrogen may accumulate within containment as a result of (1) metal-water reaction between the zirconium fuel cladding and the reactor coolant, (2) radiolytic decomposition of the water in the reactor core and containment, and (3) corrosion of metals by emergency core cooling and containment spray solutions. If a sufficient amount of hydrogen is generated, it may react with the oxygen present or generated in the containment following an accident. To monitor and control the buildup of hydrogen and oxygen within the containment, GE has provided the following systems:

(1) An Atmospheric Control System (ACS) The ACS is designed to maintain the primary containment oxygen concentration below the maximum permissible limit per Regulatory Guide 1.7 during normal, abnormal and accident conditions to assure an inert atmosphere. Inerting is accomplished with nitrogen storage tanks that are adequately sized and provided with makeup capability. The ACS is designed to withstand missiles, pipe whip, flooding, tornadoes, a safe shutdown earthquake, LOCA environment, and a single active failure. However, GE states that the ACS is non-safety grade, whereas the SRP Section 6.2.5 acceptance criteria regarding General Design Criterion 41 states that the combustible gas

control system design should be safety-grade because this system is relied on to ensure that containment integrity is maintained following an accident. Based on this discrepancy, the staff has not been able to find the requirements for design of the ACS acceptable.

(2) A Containment Atmosphere Monitoring System (CAMS) The CAMS is designed to monitor oxygen levels in the wetwell and drywell during accident conditions to confirm that the primary containment is inherited. The staff's evaluation of the CAMS is discussed in Section 7.0 of the SER.

(3) Capability of Post-LOCA Purging of the Containment Post-LOCA primary containment backup purging capability is provided in accordance with Regulatory Guide 1.7 and as an aid in containment atmosphere cleanup following a LOCA. During normal plant operation, the purge line also functions, in conjunction with the nitrogen purge line, to maintain primary containment pressure at about 0.75 psig and oxygen concentration below 4 percent by volume. This is accomplished by makeup of the required quantity of nitrogen into the primary containment through the makeup line or relieving pressure through the purge line. Flow through the bleed line will be directed through either the SGTS or the secondary containment HVAC and will be monitored for radiation release. However, GE has provided neither the purge rate that would be required to maintain the oxygen concentration below 4 percent by volume nor the radioactive consequences analysis for the staff to review. Based on its review, the staff has not been able to find the ABWR capability of post- LOCA backup purging of the containment acceptable.

(4) Penetrations for Portable Recombiners GE states that provisions are made for connection of two portable recombiners immediately after a LOCA for flammability control during accident conditions. However, GE has not discussed the availability of the hydrogen recombiners following a LOCA. Therefore, the staff has not been able to find that the use of portable recombiners is acceptable.

With respect to the post-accident hydrogen generation analysis, GE indicates that the analytical model described in GE report, NEDO-22155, "Generation and Mitigation of Combustible Gas Mixtures in Inerted BWR Mark I Containment," was used to compute the hydrogen and oxygen generation from radiolysis. The NEDO-22155 report is being reviewed by the staff for the EPRI requirements document certification. The staff will report its finding of this issue in a supplement to this SER.

GE has not provided information with respect to ABWR combustible gas control system as requested by the staff in its RAI dated July 7, 1988. The staff will evaluate the information when available and will report the findings in a supplement to this SER.

6.2.6 Severe Accident Considerations

The GE APWR containment relative to severe accidents is currently being reviewed by the staff. The evaluation will be reported in a supplement to this SER.

9.2.9 Makeup Water System (Condensate)

The makeup water condensate (MUWC) system (the condensate storage and transfer system) was reviewed in accordance with SRP Section 9.2.6, "Condensate Storage Facilities." The entire MUWC is within the scope of the ABWR.

The function of the MUWC is to provide a source of condensate quality water and a piping distribution system from the source to the components that require this water during normal and emergency operations. A 557,000 gallon condensate storage tank (CST) located outdoors adjacent to the turbine building is the source of water for this system. The CST reserves 150,000 gallons of this capacity for decay heat removal by the AC independent RCIC system for eight hours following a station blackout. The CST is of concrete construction, with a stainless steel lining. The CST also serves as the surge volume for the condensate system. Level sensing instruments and transmitters automatically switch over the high pressure core flooder (HPCF) and reactor core isolation cooling (RCIC) pumps from the preferred CST to the safety-related suppression pool when the CST level is low. The tank is also a suction source for the control rod drive supply pump (the preferred water source being the condensate treatment system) and the suppression pool cleanup (SPCU) pump which is used for filling the fuel pool makeup line when required. The MUWC system normally provides water via three system transfer pumps for charging, flushing, pump sealing, surveillance testing, room decontamination, and makeup as appropriate, for a number of systems including RHR, HPCF, RCIC, fuel pool skimmer surge tanks and the main condenser hotwell.

The MUWC system is not safety related except as noted below because it does not affect the reactor coolant system pressure boundary, capability to achieve and maintain safe shutdown, or the capability to prevent or mitigate the consequences of accidents which could result in unacceptable offsite radiological exposures. Therefore, compliance with GDCs 44 "Cooling Water," 45 "Inspection of Cooling Water System," and 46 "Testing of Cooling Water System" identified as acceptance criteria in SRP 9.2.6, is not applicable to the system. Also, compliance with GDC 5 "Sharing of Structures, Systems, and Components" identified as an acceptance criterion in SRP 9.2.6, is not applicable to the system, since the ABWR design is limited to a single unit. Although it is not safety related, this system is designed to provide adequate pump flow, CST overflow/drainage diversion to the radwaste system, material corrosion resistance, CST water level control room instrumentation, outdoor piping freezing protection, and to allow testing of air-operated valves.

As stated in the ABWR SSAR Table 3.2-1 "Classification Summary" for the MUWC, RCIC and HPCF systems, certain parts of MUWC system piping including supports and valves are designed to seismic Category I and Quality Group B standards and are located in seismic Category I, flood and tornado-missile protected structures. The safety-related portions include those forming part of the containment boundary as well as system piping portions that interface with the safety-related RCIC and HPCF systems including the isolation/suction valves for the systems from the CST. The nonsafety-related portions of the system which could affect any structures systems or components important to safety due to their failures during a seismic event are designed to assure their integrity under seismic loading resulting from a safe shutdown earthquake. The level instruments that facilitate the automatic switch over of the HPCF and RCIC pumps suction from the CST to the suppression pool and their power supplies are safety-related. However, it is not clear whether these are attached to safety-related portions of system piping, as required. Subject to meeting the above requirement for the automatic switch over devices, the system meets Positions C.1 and C.2 of Regulatory Guide 1.29 for its safety and nonsafety-related portions, respectively. GE should also verify whether an automatic switch over (to the suppression pool) feature is provided for the nonsafety- related suppression pool cleanup pump suction as stated in its submittal dated March 7, 1989.

GE has not provided the analysis for potential flooding resulting from possible failure of the nonsafety-related MUWC system including the CST and how safety-related structures, systems and components are protected from such flooding. Subject to satisfactory resolution of this concern, the system meets the requirements of GDC 2.

The specified available water volume of 150,000 gallons of the CST (the suppression pool being the other source) for a station blackout event is somewhat lower than the staff estimated volume of about 155,000 gallons, taking into account an additional water requirement for total leakage from the reactor coolant system pressure boundary (leakage for 8 hours at a rate of 25 gpm). Since the ABWR design provides for reactor internal pumps for coolant recirculation and these are located inside the reactor pressure vessel, the staff has not included any reactor coolant pump seal leakage in the above estimate of water requirements for the station blackout event. Subject to increase of the specified volume of available CST water inventory for the station blackout event as indicated above, the MUWC system complies with the applicable guidance of Regulatory Guide 1.155 "Station Blackout."

As discussed above, the MUWC system complies with Positions C.1 and C.2 of Regulatory Guide 1.29, GDC 2 and applicable guidance of Regulatory Guide 1.155 and, therefore, with the applicable acceptance criteria of SRP 9.2.6, subject to resolution of all the concerns identified above.

9.2.10 Makeup Water System (Purified)

The makeup water system purified (MUWP) was reviewed in accordance with SRP Section 9.2.3 "Demineralized Water Makeup System." The portion of the MUWP which is inside the reactor building is within the scope of the ABWR. The portions of the system which involve demineralized makeup water preparation, storage, and transport to the reactor building are within the scope of the referencing applicants. The MUWP including the purified or demineralized water storage tank is not safety related, except as noted below.

The function of the MUWP is to provide a source of demineralized makeup quality water and a piping distribution system from the source to the components that require this water. A purified water storage tank, which is not part of the GE scope of the ABWR, is the source of demineralized water for this system. The MUWP normally provides demineralized water via two system transfer pumps at a maximum flow rate of approximately 600 gpm and within a temperature range of 50 -100 F for flushing, sealing, surveillance testing, area decontamination, sampling and makeup as appropriate. A number of systems are supplied makeup water, which include the makeup water condensate (MUWE) system, reactor building cooling water (RCW) system, turbine building cooling water system, diesel generator cooling water (DGCW) system, liquid radwaste system, standby liquid control (SLC) system, and other plant auxiliary systems. Protection from flooding for safety-related structures, systems and components resulting from failure of the MUWP system is discussed in Section 3.4.1 of this DSER.

The MUWP system is not safety related except as noted below because it does not affect the reactor coolant system pressure boundary, capability to achieve and maintain safe shutdown, or the capability to prevent or mitigate the consequences of accidents which could result in unacceptable offsite radiological exposures. The MUWP line enters the primary containment through one penetration. The system piping through this penetration has a locked closed manual valve outside the containment and a check valve inside the containment. The portions of the system penetrating the containment including the above valves are designed to seismic Category I requirements in accordance with Position C.1 of Regulatory Guide 1.29.

Although it is not safety related, this system is designed to prevent any radioactivity contamination of the purified water. GE will provide additional system design features identified below which fall within its design scope. The referencing applicant will be required to provide the remaining design features. These features include testing capability for air operated valves, adequate pump NPSH, purified water storage tank overflow/drainage diversion to the radwaste system, material corrosion resistance, adequate distribution piping, valves, instruments and controls, purified water storage tank water level control room instrumentation, outdoor piping freeze protection and adequate diking and other means to control spill and leakage from the demineralized water storage tank which will be located outdoors. Table 9.2-2a of the ABWR SSAR presents chemistry requirements for the purified makeup water.

It is not clear whether:

(1) The portions of the system which interface with all safety-related (e.g., RCW, DGCW, SLC) systems, including the system isolation valves from the safety-related systems, are all safety-related and located in seismic Category I, tornado-missile and flood protected structures,

(2) Isolation of the MUWP system from safety-related systems during a LOCA is automatic, and,

(3) The nonsafety-related portions of the system, which due to their failure during a seismic event can adversely impact structures, systems or components important to safety, are designed to assure their integrity under seismic loading resulting from a safe shutdown earthquake.

Subject to verification that the design of the MUWP system meets the above requirements, the system complies with Positions C.1 and C.2 of Regulatory Guide 1.29 for the safety- and nonsafety-related portions of the system and GDC 2 for protection against natural phenomena.

GE has not specified the demineralized water makeup system and storage tank capacities required to meet all the normal operating requirements of demineralized water. This must be identified as an interface requirement. Also, GE has not indicated the system provisions to prevent (1) radioactivity intrusion into the system from other potentially radioactive systems with which it interacts, and (2) supply of out-of-specification water to safety-related systems. Further, it is not clear whether the portions of the MUWP system in buildings other than the reactor building (e.g., turbine building) and transport of the demineralized water to these buildings are within GE's scope or the referencing applicant's scope. The staff requires GE to address all the above issues to complete the information on the MUWP system provided in ABWR SSAR Section 9.2.10.

As discussed above, the MUWP system complies with Positions C.1 and C.2 of Regulatory Guide 1.29 and with GDC 2 and, therefore, with the applicable acceptance criteria of SRP 9.2.3, subject to resolution of the concerns identified above relating to the design of safety- and nonsafety-related portions of the system.

9.2.11 Reactor Building Cooling Water System

The reactor building cooling water (RCW) System was reviewed in accordance with SRP Section 9.2.2 "Reactor Auxiliary Cooling Water Systems." The function of the RCW system is to remove heat from plant auxiliaries, some of which are required for safe shutdown and following a LOCA. The RCW is required to operate at normal power, reactor shutdown, hot standby, both with and without preferred AC power available, and after a postulated LOCA has occurred. The RCW is a closed cooling water system which provides cooling water to the following essential systems and components: residual heat removal (RHR) and fuel pool cooling (FPC) heat exchangers; mechanical seals and motor bearings for RHR and high pressure core flooder (HPCF) pumps; air conditioning units for pump rooms (RHR, HPCF, FPC, and reactor core isolation cooling (RCIC)) and system rooms (standby gas treatment, containment atmospheric monitoring, and flammability control); inner coolers and filtered water and lubricating oil coolers for diesel generators; and HVAC emergency cooling water (HECW) system refrigerators. Additionally, the RCW supplies cooling water to the nonessential reactor internal pump (RIP) motor coolers and motor generator sets, drywell coolers, reactor water cleanup (RWCU) pump coolers, instrument and service air (IA and SA) system coolers, RWCU non-regenerative heat exchangers, control rod drive (CRD) pump oil coolers, and other nonessential auxiliary components in the reactor, turbine and radwaste buildings (e.g., radwaste components, condenser offgas, and reactor building and turbine building sampling coolers).

The RCW system supplies cooling water which picks up heat from the plant auxiliaries it serves and rejects the heat through the RCW heat exchangers to the reactor service water (RSW) system. The RSW, in turn, rejects the heat to an ultimate heat sink that will be designed by the individual applicants referencing the ABWR. The GE scope of the RCW system includes all the piping, valves, pumps, heat exchangers, instrumentation, and controls from the RCW heat exchangers to their loads in the reactor, turbine, and radwaste buildings. GE has specified the total heat removal rate, total flow rate, temperature drop and pressure drop at the RCW heat exchangers for all modes of operation identified above. These parameters are the interface requirements for the referencing applicant to design the plant-specific ultimate heat sink system which would be connected to the RSW system. Additionally, GE has specified the inlet and outlet temperatures during applicable modes of operation for components serviced by RCW.

The RCW system is composed of three mechanically and electrically independent divisions. Each division consists of its own separate piping (including supply and return headers), two pumps, two heat exchangers, valves, and instrumentation. Each RCW division is powered by a different division of the ESF power system. Each RCW division supplies cooling water to the auxiliaries of a separate emergency diesel generator, RHR heat exchanger, RHR pump room air conditioning unit and RHR pump motor and seal coolers. Other safety loads and non-essential cooling loads are distributed among the three divisions with two divisions sharing the loads for systems with redundant components (e.g., HPCF, HECW, SGTS). Each division has one isolable train for non-essential loads.

Each RCW division is equipped with a surge tank which GE states is designed to accommodate thirty days of system design leakage without makeup (response dated March 7, 1989). Also, the system is designed to detect system leakage by associated level monitors, provide adequate pressure for pump suction, and allow for changes in system water volume without significant pressure variations. The system is initially filled with demineralized water from the makeup water (purified) system. Each division is further equipped with a chemical addition tank to add chemicals to the RCW to protect it from corrosion or organic fouling.

System protection from water hammer is achieved by the use of high point vents in isolable portions of the system and operational procedures requiring filling and venting of any sections of the system prior to operation. Non-essential RCW system cooling loads are automatically isolated by applicable valve closure in the event of a LOCA with the exception of system cooling loads to IA and SA system coolers, CRD pump oil coolers and RWCU pump coolers; these are isolated by operator, if so desired. Level switches provided for the surge tank facilitate automatic isolation of non-essential cooling loads in the event of significant system leak resulting from piping failures in the nonsafety-related portions of the system. One valve on each supply and discharge line, with suitable power and controls from applicable divisional sources assures isolation in the event of a single active component failure. However, this will not affect supply of cooling water to essential cooling loads from other divisions. GE has described the detailed procedures for determining whether the system leakage occurs in the non-essential portion of the system. A falling surge tank level would detect such leakage. Radiation monitors located downstream of the RCW pumps and heat exchangers facilitate detection that radiation has leaked into that division. The ABWR provides remote manual isolation capability for any division. The two remaining operable divisions will be sufficient to meet the total essential cooling load.

The RCW system is comprised of safety and nonsafety-related portions. Portions of the system piping including valves forming part of the primary containment boundary and other safety-related portions of the system piping up to and including the isolation valves which isolate the system from its nonsafety- related portions are designed to seismic Category I, Quality Group B or C and 10 CFR 50, Appendix B requirements. The safety-related portions include the RCW pumps, heat exchangers, surge tanks and the division isolation valves. Instrumentation and controls performing safety-related functions (e.g., surge tank level switches) are located in the safety-related portions of the system. Electric modules (e.g., sensors, power supplies, signal processors) and cables performing safety-related functions are all designed to seismic Category I and Quality Assurance B requirements. Nonsafety-related portions of the system that can adversely impact safety-related structures, systems or components due to their possible failure during a seismic event are designed to ensure their integrity under seismic loading resulting from an SSE. The safety-related portions are located in seismic Category

I, flood and tornado-missile protected structures; GE shall provide confirmatory documentation that this includes also safety-related electric modules and safety-related cables as they indicated in a telephone conversation with the staff in November 1989. Based on the above, the staff finds that the design of the RCW system complies with GDC 2 with respect to its protection from natural phenomena, and meets Positions C.1 and C.2 of Regulatory Guide 1.29 with respect to its seismic requirements for the safety and nonsafety-related portions.

GE stated that both the mechanical equipment and piping and electrical equipment including instrumentation and controls of the redundant divisions of the RCW system are sufficiently separated and protected to ensure availability of the needed equipment to perform reactor shutdown in the event of any of the following occurrences: pipe rupture or equipment failure induced flooding, spraying or steam release; pipe whip and jet forces from a postulated nearby high energy pipe line break; missiles from equipment failure; fire; non- Category I equipment failure; or a single active component failure in the system. GE has performed a failure analysis of the RCW system and presented the results in the ABWR SSAR to demonstrate that a single active or applicable passive component failure will not compromise the ability of the RCW system to transfer heat loads from safety related components to the reactor service water system under all applicable modes of operation. Also, GE has provided design characteristics for RCW system components (e.g., pump design flow rate; heat exchanger heat removal capacity) to show that the system is capable of transferring the expected heat loads to the reactor service water under all operating conditions.

Based on its review of the above, the staff has identified the following concerns:

(1) ABWR SSAR Section 3.5.1 does not include the RCW system (safety-related portions) in the list of systems requiring missile protection. Also, GE has not explained how the safety-related portions of the RCW system are protected against missiles generated by nonsafety-related components. In Amendment No. 7 (see response to Question No. 410.5), GE stated that safety-related divisions of the system are physically separated from each other. The staff notes that the above design feature does not protect safety-related portions from being impacted adversely by missiles generated by any nonsafety-related components.

(2) It is not clear whether the heat removal capacity of the system heat exchangers includes an appropriate allowance for reduction in capacity due to fouling of the heat exchangers.

(3) A RWC heat exchanger heat removal design capacity based on heat load required to be removed during suppression pool cooling following a LOCA with the pool at 97 C may be inadequate. A reactor shutdown at 4 hours after a blowdown

to the main condenser may be the bounding case. This may require a greater heat removal rate and consequently, a higher design capacity than that currently stated for the heat exchangers (see GE's response to Question No. 440.73).

(4) The projected heat loads and flow rates for hot standby conditions with a loss of AC power indicate that both heat exchangers and both RCW pumps in a division are required. This will be the case for shutdown at 4 hours also. The current SSAR write up should be revised to incorporate the above requirement.

(5) GE has not addressed the specific provisions to protect the safety-related portions of the system against the effects of postulated high-energy and moderate-energy line failures.

(6) It is not clear whether the loss of an RCW division during normal operation will result in plant shutdown or operation at reduced power. (For example, the loss of Division A which supplies cooling water to 5 out of 10 RIP coolers and 2 out of 3 drywell coolers).

(7) Piping and instrumentation diagrams for the RCW system show the interfacing reactor sea water system (presumably, this has to be corrected to depict a service water system). However, there is neither a description of the system, nor an explicit interface requirement for this system in the SSAR, if it is within the scope of the referencing applicant.

Subject to resolution of the above concerns, the safety-related portions of the RCW system comply with GDC 4 with respect to protection against dynamic effects resulting from postulated piping failures and internally and externally generated missiles and with GDC 44 with respect to the provisions of a system to transfer heat from structures, systems and components important to safety to an ultimate heat sink.

All three divisions of the RCW system will have at least one RCW pump operating. This configuration ensures the immediate availability of the RCW system for plant shutdown in the event of a LOCA. A loss of offsite power concurrent with a LOCA will result in a temporary loss of pumping until the automatically sequenced restart of RCW pumps from the emergency diesel generator loading sequence. Upon the occurrence of a LOCA, most non-safety RCW loads are automatically isolated as stated above, the second RCW pump started and second heat exchanger in each division is placed in service.

As stated earlier, GE has provided interfacing requirements for an applicant to design the ultimate heat sink in terms of the total heat load, temperature drop, pressure drop, and flow through the RCW heat exchangers. The RCW system water quality requirements are established by makeup water system (purified), since this system provides the makeup water for the RCW system.

This system is discussed separately under Section 9.2.10 of this DSER.

All three divisions of the RCW system are designed to allow periodic inservice inspection of all the system components. This testing capability consists of structural and leaktightness visual inspection, entire system operability, and system component operability and performance. Testing will be conducted to simulate as closely as possible the entire operational sequence of the RCW system for reactor shutdown and LOCA. The system design also incorporates provisions for accessibility to permit inservice inspection as required. Based on the above, the staff finds that the system complies with GDC 45 and GDC 46 with respect to inspection and testing requirements for cooling water systems.

Based on the above, the staff concludes that the design of the RCW system complies with GDCs 45, 46 and 2 with respect to inservice inspection and testing requirements and protection against natural phenomena for its safety-related portions. The staff also concludes that the system design meets the guidelines of Positions C.1 and C.2 of Regulatory Guide 1.29 with respect to seismic requirements for the safety-related and applicable nonsafety-related portions of the system. Further, the staff concludes that the system design complies with GDCs 44 and 4 with respect to cooling water requirements and protection against internally and externally generated missiles and dynamic effects resulting from postulated piping failures, subject to satisfactory resolution of all the concerns identified above. The system design meets the applicable acceptance criteria of SRP 9.2.2 subject to satisfactory resolution of all the concerns identified above.

9.2.12 HVAC Normal Cooling Water System

The HVAC normal cooling water (HNCW) system was reviewed in accordance with SRP Section 9.2.2. The entire HNCW system is within the scope of the ABWR. With the exception of portions of the system which penetrate the primary containment, the portions of the system which are part of the secondary containment boundary, and the associated isolation valves, the HNCW system is not safety related.

The function of the HNCW system is to provide chilled water to the drywell cooler cooling coils and cooling coils of other non-safety-related air conditioners, primarily, in reactor, control, radwaste, and service buildings. The HNCW system is not safety related because it is not required to assure the RCS pressure boundary, capability to achieve and maintain safe shutdown, and the ability to prevent or mitigate offsite radiological exposures during accidents. Therefore, GDCs 44, 45, and 46 identified as acceptance criteria in SRP 9.2.2 for safety-related portions of cooling water systems are not applicable to the HNCW system. The HNCW system interfaces the primary containment through two penetrations: one for the supply line and the other for the return line. The supply line penetration has one motor operated isolation valve outside the containment and a check (isolation) valve inside the containment. The return line penetration has two motor operated isolation valves, one inside and one outside the containment. Isolation valves and piping for the primary containment penetrations are safety-related and are designed to Seismic Category I, Quality Group B, and 10 CFR 50, Appendix B standards. Piping for penetrations for secondary containment is designed to seismic Category I and 10 CFR 50, Appendix B standards. GE shall confirm that the above requirements are included for HNCW system isolation valves for secondary containment penetrations.

The rest of the system is nonsafety-related as stated above and is designed to non-seismic Category I standards. However, the nonsafety-related portions of the system whose failure during a seismic event could affect any structure, system or component important to safety, are designed to assure their integrity under seismic loadings resulting from a safe shutdown earthquake. Based on the above, the staff finds that the design of the HNCW system meets Positions C.1 and C.2 of Regulatory Guide 1.29 as addressed by the SRP 9.2.2 acceptance criterion with respect to the seismic requirements for the safety and nonsafety-related portions of the system. This finding is subject to GE's confirmation that the safety-related portions include the isolation valves for the secondary containment penetrations. By virtue of their location in seismic Category I, tornado-missile and flood protected structures, the safety-related portions of the system are protected against adverse natural phenomena. Further, all safety-related systems are protected against flooding that may result in the event of system failure as concluded in Section 3.4.1 of this DSER. Based on the above, the staff finds that the system complies with GDC 2 with respect to protection of its safety-related portions against natural phenomena and protection of other safety-related systems against the consequences of failure of the non-seismic portions of the system, as required by SRP 9.2.2 acceptance criterion.

Although it is not safety related, this system is designed to allow periodic testing and inspection of major components. Appropriate ASHRAE, ASME, TEMA, and Hydraulic Institute standards are used for all tests. The major components of the HNCW system consist of five 25% capacity chillers (one standby) each with an HNCW pump (one standby) and the associated piping, valves, and instrumentation. The system is also provided with a chemical feed tank and vents at high points, the latter to eliminate water hammer. Cooling water to the chiller-condenser is supplied by the turbine building cooling water (TCW) system.

Makeup water to the system is supplied by the TCW system surge tank which, in turn, receives water from the MUWP system. The MUWP and the TCW systems are evaluated in Sections 9.2.10 and 9.2.14 of this SER. GE has provided the design characteristics for the system (e.g., cooling capacity of the chillers, pump design flow rate, chilled water supply temperature) and the heat loads required to be removed from the components served by the system. The above characteristics indicate that the system is capable of meeting the cooling water needs of the components it serves during normal plant operation and refueling shutdown. The chiller units are controlled individually by remote manual switches. The containment isolation valves for the system close automatically on a LOCA signal. These valves can be also operated remote-manually. The piping and instrumentation diagram (ABWR SSAR Figure 9.2-2a) for the system shows only four chillers and four pumps, whereas ABWR SSAR Section 9.2.12 and Table 9.2-6 indicate five chillers and five pumps. GE should correct this discrepancy. Based on the above, the staff concludes that the HNCW system meets the applicable acceptance criteria of SRP 9.2.2, subject to correction of the discrepancy mentioned above, and confirmation relating to the seismic design of the system isolation valves for the secondary containment penetrations.

9.2.13 HVAC Emergency Cooling Water System

The HVAC emergency cooling water (HECW) system is a closed cooling water system that was reviewed in accordance with SRP Section 9.2.2. The function of the HECW system is to provide cooling water to the main control room air conditioners, diesel generator zone coolers, and control building essential electrical equipment room coolers. The HECW system is required to operate at normal power, reactor shutdown, and after any postulated abnormal reactor conditions including a LOCA. The HECW system has no primary or secondary containment penetrations. GE states that the entire HECW system is safety- related. The entire system is within the scope of the GE ABWR design. Specifically, the GE scope of the HECW system includes all piping, valves, pumps, chillers, instrumentation, and controls from the HECW chillers to their cooling loads. There is no interface requirement for referencing applicants for this system because it interfaces with systems that are also part of the GE scope of the ABWR design (e.g., the RCW and MUWP systems, which are discussed in Sections 9.2.11 and 9.2.10 of this DSER).

The HECW system is comprised of two mechanically and electrically independent and completely redundant divisions. Each division, with a surge tank, two 50% capacity chillers (refrigeration units) and two 100% capacity pumps, consists of its own separate piping (including supply and return headers), valves, and instrumentation. Each refrigeration unit includes a condenser, an evaporator, a centrifugal compressor, refrigerant, refrigerant piping and package chiller controls. Cooling water to the condensers is supplied by the corresponding RCW system divisions. Each HECW division is powered by a different division of the ESF power system. Divisions A and B supply chilled water to cooling coils in the essential electric equipment rooms A and B and diesel generator Zones A and B respectively. Both the divisions supply chilled water to their respective coolers in the main control room. The system is also provided with a chemical feed tank to add chemicals to each division to protect the system components from fouling.

The HECW system, as well as the cooling water lines from the RCW system, is designed to Seismic Category I, 10 CFR 50 Appendix B, and Quality Group C requirements. Thus, the system meets Position C. 1 of Regulatory Guide 1.29 with regard to seismic classification for safety-related systems. By virtue of its location in scismic Category I, flood and tornado-missile protected structures, the system complies with GDC 2 with regard to protection of safety-related systems against adverse natural phenomena. In this context, the staff notes that location RZ (reactor building clean zone) has not been included under the "Location" column in Table 3.2-1, page 3.2-22 of the ABWR SSAR for the safety-related portions of the HVAC cooling water systems. Since the HECW system contains a portion (diesel generator area coolers) in the above location, the table needs revision to include location RZ.

Each HECW division is equipped with a surge tank which GE states is designed to accommodate over 100 days system leakage without makeup during an emergency. The surge tank is connected to the MUWP system which provides normal makeup. The tank includes level switches to detect system leakage and to facilitate supply of makeup water to the tank when required. These switches actuate the makeup water supply valves (open or close on low or high tank water level) and provide annunication of control room alarms (high-high or low-low tank water levels). (See ABWR SSAR Section 7.3.1.1.9).

The design of the HECW system includes sufficient separation for both mechanical and electrical components of the redundant trains and protection for the system to perform its function under all reactor conditions including LOCA, loss of normal AC power, or a single active component failure in the system or any combination of the above. GE performed a failure analysis of the HECW system and presented the results in the ABWR SSAR to demonstrate that failure of a single active component, failure of all power to a single class 1E power system bus or failure of refrigerator signal will not compromise the ability of the system to perform its function. With the system controls set for automatic operation, the system is automatically initiated whenever the HVAC systems in the control building or diesel generator areas are started. The system can also be manually started from the control room. Interlocks provided for the chillers automatically start the redundant division whenever there is failure of the operating division (e.g., high temperature of the returned cooling water, inadequate chilled water flow). The system flow switches prevent the chiller from operating unless there is sufficient water flow through both the evaporator and the condenser. The chiller units can be controlled individually from the control room by remote-manual switches. The system includes instrumentation and controls for monitoring and controlling system parameters, such as chilled water flow and temperature, condenser water flow, and evaporator discharge flow and temperature. Since the system is not expected to contain any significant level of radioactivity, it has no radiation monitors. GE has provided the design characteristics for the system components (e.g., capacity of the HECW system refrigeration units, chilled water pump flow rate, chilled water and condenser water supply temperatures). GE has also provided the heat removal and flow requirements for the individual system components. This information indicates that any single division of the system is by itself capable of rejecting the total heat from the components the system serves via the refrigerant to the RBCW cooling water under all reactor conditions.

Based on its review of the above, the staff has identified the following concerns:

(1) The ABWR SSAR Section 3.5.1 does not include the HECW system in the list of systems requiring missile protection. Also, GE has not explained how the safety-related HECW system divisions are protected from missiles generated by nonsafety-related components. In Amendment No. 7 (response to Question No. 410.5), GE has stated that safety-related divisions of systems are physically separated from each other. The staff notes that the above design feature does not protect safety-related divisions from being impacted adversely by missiles generated by nonsafety-related components.

(2) GE has not identified the system design features provided to preclude the adverse effects of potential water hammer.

(3) GE has not listed the specific provisions provided to protect the safety-related divisions against the effects of postulated high-energy and moderate energy line failures.

(4) ABWR SSAR Table 9.2-9 does not indicate that a single HECW pump by itself can deliver the required total chilled water flow rate. Therefore, the statement in ABWR SSAR Section 9.2.13.2 that each division of the HECW system contains two 100 percent capacity pumps has to be corrected.

(5) It is not clear whether the single chemical feed tank provided for the system is safety-related. Also, GE has not indicated whether the associated isolation valve and the portion of the piping wherein it is located are safety-related. Further, GE has not indicated how the nonsafety-related portions of the HECW system (if there are any such portions) is isolated from the rest of the system whenever such isolation is warranted.

(6) It is not clear which division of the HECW system supplies chilled water to the cooling coils in the diesel generator Zone C. Also, ABWR SSAR Figure 9.4-4 indicates three

divisions for the HECW system whereas ABWR SSAR Section 9.2.13 refers to only two divisions for the system. The staff considers that the Division C diesel generator support systems should be independent of Divisions A and B.

(7) The piping and instrumentation diagrams for the HECW system identify three sheets; but only two have been provided.

Subject to resolution of the above concerns, the system complies with GDC 4 with respect to protection for the system against dynamic effects resulting from postulated piping failures and internally and externally generated missiles and GDC 44 with respect to the applicable requirements for this cooling water system.

As discussed above, the only interfacing requirements for this system are the HECW chiller condenser flow, supply temperature, and heat removal capacity which are directly input to the GE designed RCW system. No applicant interface requirements are necessary for the HECW system. The HECW system water quality requirements are established by the makeup water system (purified) since this system provides the water for the HECW system surge tanks.

Both divisions of the HECW system include provisions to allow periodic inservice inspection of all the system components to ensure the integrity of the system and its capability to perform its intended function. Local display devices are provided to indicate vital parameters required in testing and inspections. For example, system chilled water flow rate and temperature can be checked out by readout of locally mounted pressure and temperature gauges at the main control panel. In a telephone conversation with the staff on February 26, 1990, GE states that the system also includes provisions to permit periodic testing of system components as well as the system as a whole. Specifically, GE stated that this testing capability would include structural and leaktightness visual inspection, entire system operability, and system component operability and performance. Based on the above, the staff finds that the system complies with GDC 45 and GDC 46 with respect to the requirements for inspection and testing of safety-related cooling water systems contingent upon GE updating the ABWR SSAR reflecting what was stated in the above telephone conversation.

Based on the above, the staff concludes that the design of the ABWR HECW system meets the requirements of GDC 2, 45, and 46 with respect to its protection against natural phenomena, inservice inspection, and functional testing contingent upon GE updating the ABWR SSAR reflecting what was stated in the February 26, 1990 telephone conversation. The HECW system also meets the guidelines of Position C.1 of Regulatory Guide 1.29 with respect to its seismic classification. Further, the staff concludes that the system design complies with GDCs 4 and 44 with respect to protection against internally and externally generated missiles and dynamic effects resulting from postulated piping failures, and cooling water requirements, subject to satisfactory resolution of all the concerns identified above. Also, as noted above, the staff requires the revision of the applicable entry in ABWR SSAR Table 3.2-1. The system design meets the applicable acceptance criteria of SRP 9.2.2 subject to satisfactory resolution of all the concerns identified above and appropriate revisions of Table 3.2-1 and Section 9.2-13 of the ABWR SSAR.