September 26, 1988

Docket No. STN 50-605

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

Dear Mr. Marriott:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING THE GENERAL ELECTRIC COMPANY APPLICATION FOR CERTIFICATION OF THE ABWR DESIGN

In our review of your application for certification of your Advanced Boiling Water Reactor Design, we have identified a need for additional information. Our request for additional information, contained in the enclosure, addresses the areas of SRP Chapter 3 reviewed by the Mechanical Engineering Branch. This completes the initial request for additional information on the ABWR related to SSAR Chapters 1, 2 & 3. However, the need for additional information may occur during the development of the staff's safety evaluation. If this should occur, the need will be identified in a draft Safety Evaluation Report which will be provided for your consideration.

In order for us to maintain the ABWR review schedule, we request that you provide your responses to this request by November 30, 1988. If you have any concerns regarding this request please call me on (301)492-1104.

Sincerely,

/s/
Dino C. Scaletti, Project Manager
Standardization and Non-Power
Reactor Project Directorate
Division of Reactor Projects - III. IV, V and Special Projects
Office of Nuclear Reactor Regulation

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

September 26, 1988

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Standardization and Non-Power
Reactor Project Directorate
Division of Reactor Projects - III, IV,
V and Special Projects
Office of Nuclear Reactor Regulation

Enclosure:
As stated

REQUEST FOR ADDITIONAL INFORMATION
ADVANCED BOILING WATER REACTOR STANDARD SAFETY ANALYSIS
REPORT, DOCKET NO. 50-605
MECHANICAL ENGINEERING BRANCH

210.3 In Subsection 3.1.2.1.1.2, "Evaluation Against Criterion 1", a footnote states that "important-to-safety" and "safety-related" are considered equivalent in this SSAR. The staff does not agree with this definition. The staffs' position on this issue remains as stated in NRC Generic Letter 84-01, "NRC Use of the Terms "Important to Safety" and "Safety-Related", dated January 5, 1984. The staff used this position as guidance in its reviews of applications for operating licenses of nuclear power plants for a number of years prior to the issuance of GL 84-01. During these reviews, the staffs' evaluations of the quality assurance requirements in 10 CFR Part 50, Appendix B generally applied to the narrower class of "safety-related" equipment as defined in 10 CFR Part 50.49(b)(1), 10 CFR Part 100, Appendix A and in Section 3.2 of this SSAR. This implied that normal industry practice for quality assurance was generally acceptable for most equipment not covered by the "safety related" definition. However, as pointed out in Generic Letter 84-01, there have been specific situations in the past where the staff has determined that quality assurance requirements beyond normal industry practice were needed for components and equipment in the more broad "important to safety" class.

It is the staffs' opinion that a strict interpretation of the ABWR position on this issue could result in an unacceptable classification of structures, systems and components for Table 3.2-1 in this SSAR.

Revise the footnote in Subsection 3.1.2.1.1.2 and the discussion in Section 3.2 to be consistent with the staff's position as stated in Generic Letter 84-01. It should be made clear that the staff's position will not result in a broadening of the staff's review. Rather, it provides the basis which the staff has been using and continues to use as guidance in its reviews of Quality Group Classification for certain components and equipment which are not included in the "safety-related" definition.

210.4 In Subsection 3.2.3, "Safety Classifications", ANSI/ANS 52.1 - 1983, "Nuclear Safety Criteria for the Design of Stationary BWR Plants" is referenced for the definitions of safety classes. The guidance in this document for components which are not within the scope of Regulatory Guide 1.26 has not been endorsed by the staff. Therefore, the staff does not completely accept ANSI/ANS 52.1 for the definitions of all safety classes. Questions 210.5, 210.13, 210.15, 210.17, 210.44, and 210.45 are based on this position. To assure that Table 3.2-1 will be consistent with similar tables in recently licensed BWR/6 plants, such as Perry and River Bend, the reference to ANSI/ANS 52.1 - 1983 should be either eliminated or revised.

210.5 In Table 3.2-1, Items B1.7, "Control Rods" and B1.9, "Fuel Assemblies" are classified as Safety Class 3, which is consistent with the criteria in the ANSI/ANS 52.1 - 1983 Standard. As stated in Question 210.4, the staff does not agree with all of the recommendations in that Standard. The staff position is that Control Rods and Fuel Assemblies should be Safety Class 2 and Quality Group B. To be consistent with this position and with staff reviews on recent BWR/6 plants, such as Perry and River Bend, revise Table 3.2-1 to change the classifications of the Control Rods and Fuel Assemblies from Safety Class 3 to 2 and add Quality Group B.

Questions 210.44 and 210.45 provides similar staff positions for Item B1.5 Safety-Related Reactor Internal Structures and Core Support Structures.

210.6 In Table 3.2-1, Item B2.5 identifies Main Steam Line (MSL) piping from the outermost isolation valve to and including the seismic interface restraint as being Safety Class 1 and Quality Group A. Figure 5.1-3b, "Nuclear Boiler System P&ID, Sheet 2" identifies the same portion of the MSL as Quality Group B. Beyond the seismic interface restraint, the MSL piping is Quality Group D, which is not acceptable to the staff. To be acceptable, the MSL should be classified as recommended in Standard Review Plant 3.2.2, "System Quality Group Classification", Appendix A, i.e., Quality Group B from the outermost isolation valve to the turbine stop valve. This staff position is based on the assumption that the ABWR MSL design differs from the BWR/6 design in that it does not contain a shutoff valve in addition to the two containment isolation valves. Revise Table 5.1-3b, Table 3.2-1, Subsection 3.9.3.1.3 and Subsection 5.4.9.3 to be consistent with the above staff position.

210.7 Item B2.5 in Table 3.2-1 does not appear to agree with Figure 5.1-3c, "Nuclear Boiler System P&ID, Sheet 3". Item B2.5 states that piping in the Feedwater (FW) Systems from the outermost isolation valve to and including the seismic interface restraint is Safety Class 1 and Quality Group A. Figure 5.1-3c shows the FW line as Quality Group A up to the first spring closing check valve outside containment (F262A). The FW piping is Quality Group B between valves F262A and F282A and Quality Group D beyond F282A. There does not appear to be a seismic restraint in Figure 5.1-3c. Assuming that the ABWR FW line is similar to the BWR/6 designs, i.e., valve F282A is a shutoff valve in addition to the two containment isolation valves, the Quality Group classification of this line does not appear to be consistent with the guidelines of Standard Review Plan 3.2.2, Appendix B. Revise Table 3.2-1, Figure 5.1-3c and Subsection 5.4.9.3 to be consistent with the staff position on Quality Group in SRP 3.2.2, Appendix B. The transition from Quality Group B to D should be at the seismic interface restraint rather than shutoff valve F282A.

210.8 In Table 3.2-1, Item B3.1, the primary side recirculating motor cooling system piping is classified as Safety Class 3 and Quality Group C. In Subsection 3.9.3.1.4, this piping is described as being designed to the ASME Code, Section III, Subsection NB-3600, which is comparable to Safety Class 1. In Figure 5.4-4, "Reactor Recirculation System P&ID", this piping is identified as Quality Group A. The staff's position is that this piping should be, as a minimum, Safety Class 1, Quality Group A and meet the requirements of 10 CFR 50, Appendix B from the interface of the piping with the pump motor casing to and including the first pipe support. The remainder of this piping should be as a minimum, Safety Class 2. In addition, Item B3.2, the supports for this piping, should be the same Safety Class as the supported piping. Revise Items B3.1 and B3.2 in Table 3.2-1 to be consistent with the staff position.

210.9 In Table 3.2-1, add the classification summary for the Control Rod Drive Mechanism and the Low Pressure Core Flooder System or provide a justification for not including this information. The staff position on the Safety Class of these systems is as stated in Questions 210.5 and 210.45.

210.10 Provide the basis for all Control Rod Drive System valves (Item C1.1 in Table 3.2-1) to be classified as Non-Nuclear Safety and Non-Seismic.

210.11 Provide the basis for portions of piping systems within the outermost isolation valves in the Residual Heat Removal System and the Reactor Core Isolation Cooling System (Items E1.3, E4.1, and E4.6 in Table 3.2-1) to be classified as Safety Class 2 and 3.

210.12 Items E2.1 and E2.5 in Table 3.2-1 classifies some pumps and valves within the outermost isolation valves in the High Pressure Core Flooder System as Safety Class 2. Provide the basis for this classification.

210.13 In table 3.2-1, Item F4.1, "Refueling Equipment Platform Assembly" is classified as Non-Nuclear Safety. To be consistent with the staff position as stated in Question 210.4 and with staff reviews on recent BWR/6 plants, such as Perry and River Bend, revise Table 3.2-1 to change this classification to Safety Class 2 and Quality Group B.

210.14 If a Fuel Transfer System or Tube is applicable to the ABWR, add the Classification Summary for this system under Item F4, "Refueling Equipment" of Table 3.2-1.

210.15 In Table 3.2-1, Items F5.1, "Fuel Storage Racks-New and Spent" and F5.2, "Defective Fuel Storage Container" are classified as Non-Nuclear Safety. Item F5.2 is also classified as Non-Seismic. To be consistent with the staff position as stated in Question 210.4 and with staff reviews on recent BWR/6 plants, such as Perry and River Bend, revise Table 3.2-1 to change the classification of Items F5.1 and F5.2 to Safety Class 3 and Quality Group C. In addition, change the seismic classification of Item F5.2 to Seismic Category I and add "B" in the Quality Assurance column for F5.2.

210.16 In Table 3.2-1, the following components in the Reactor Water Cleanup System are correctly classified as Quality Group C, but are also classified as Non-Nuclear Safety:

G1.1 - Vessels

G1.2 - Regenerative Heat Exchanges

G1.3 - Cleanup Recirculation Pump

G1.5 - Pump suction and discharge piping beyond containment isolation valves.

G1.8 - Non-regenerative heat exchanger tube inside and piping and valves carrying process water.

G1.11 - Filter demineralizer holding pumps, valves and piping.

To be consistent with the discussions in Subsections 3.2.2 and 3.2.3 and with the information in Tables 3.2-2 and 3.2-3, the staff is of the opinion that all of the above components should be classified as Safety Class 3 in addition to Quality Group C. Revise Table 3.2-1, Items G1.1, G1.2, G1.3, G1.5, G1.8, and G1.11 to change the Safety Class from "N" to "3" or provide a justification for not doing so.

210.17 In Table 3.2-1, Items G2.3, "Heat Exchangers", G2.4, "Pumps and Pump Motors", G2.5, "Piping, Valves", and G2.7 "RHR Connections" in the Fuel Pool Cooling and Cleanup System are all classified as Non-

Nuclear Safety, which is consistent with the criteria in the ANSI/ ANS 52.1 - 1983 Standard. As stated in Question 210.4, the staff does not agree with all of the recommendations in that Standard. The staff position is that all of the above items should be Safety Class 3, Seismic Category 1 and listed under Quality Assurance requirements of 10 CFR 50, Appendix B. Regulatory Positions C.2 in Regulatory Guide 1.26 and C.1 in Regulatory Guide 1.29 includes this position. To be consistent with this position and with staff reviews on recent BWR/6 plants, such as Perry and River Bend, revise Table 3.2-1 to change the classification of Items G2.3, G2.4, G2.5, and G2.7 from Non-Nuclear Safety to Safety Class 3, add Seismic Category 1 and add "B" under Quality Assurance Requirement.

210.18 A staff position is that piping and valves forming part of primary containment boundary should be Seismic Category 1. In table 3.2-1, piping and valves in the Reactor Building Cooling Water System which form part of the primary containment boundary are classified as Non-Seismic. Revise Table 3.2-1 to add Seismic Category 1 to the classification of Item P2.1 or provide a justification for not doing so.

210.19 In Table 3.2-1, the following items are classified as Seismic Category 1 without a commitment to the Quality Assurance Requirement:

B3.1 - Reactor Recirculation System piping, primary side, motor cooling.

F4.1 - Refueling equipment platform assembly.

F5.1 - Fuel storage racks, new and spent.

The staff position, as discussed in Position C.1 and C.4 of Regulatory Guide 1.29 is that quality assurance requirements of 10 CFR 50, Appendix B should be applied to all structures, systems and components which are classified as Seismic Category 1. Revise Table 3.2-1 to add "B" in the Quality Assurance Requirement column for Item B3.1, F4.1, and F5.1.

210.20 One of the staff positions relative to component supports is that the Safety Class, Quality Group, Quality Assurance and Seismic Category classifications shall be identical for the supports and the supported component. Provide a commitment to this position in Table 3.2-1 and, if applicable, in Subsection 3.9.3.4, "Component Supports".

210.21 In Subsection 5.2.1.1, Table 3.2-4 is referenced to show the ABWR compliance with the rules of 10 CFR 50, Codes and Standards. Subsection 3.2 in the SSAR does not contain a reference to Table 3.2-4. In either Subsection 3.2 or 5.2.1.1, provide the information requested in Standard Review Plan, Section 5.2.1.1, "Compliance With the Codes and Standards Rule, 10 CFR 50.55a". This information should include the component Code, Code Edition and Code Addenda which will be applicable to ABWR pressure vessels, piping, pumps, valves, tanks, component supports and equipment.

210.22 Regulatory Guide 1.151 "Instrument Sensing Lines", dated July, 1983 conditionally endorses the Instrument Society of America Standard ISA-S67.02, "Nuclear Safety-Related Instrument Sensing Line Piping and Tubing Standards for Use in Nuclear Power Plants," 1980 as a basis acceptable to the NRC staff for the design and installation of safety-related instrument sensing lines in nuclear power plants. In addition to the commitment in Table 1.8-20, provide a statement in either Section 3.2 or 3.9 of the SSAR, that the design of safety-related instrument lines for the ABWR will be in conformance with Regulatory Guide 1.151. Footnote g to Table 3.2-1 is related to this issue, but does not provide an explicit commitment to R.G. 1.151.

210.23 Subsection 3.6.1.1.3(2) states that a pipe break event will not occur simultaneously with a seismic event. This does not agree with Standard Review Plan, Section 3.6.1, Branch Technical Position ASB 3-1, Paragraph B.2.b(1) or with the staffs' interpretation of Plant Event 8 in Table 3.9-2 of the SSAR. Revise Section 3.6.1.1.3(2) to be consistent with the staff position in SRP 3.6.1 or provide a justification for not doing so.

210.24 The discussion in Subsection 3.6.2.2.1 (a) through (e) relative to the methodology used to determine blowdown forcing functions requires more detailed information. Either revise this subsection to provide a commitment to the non-mandatory Appendix B of ANS 58.2, "Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Ruptures", or provide the following:

a. Provide a detailed discussion of the basis for the 0.7 thrust coefficient in Subsection 3.6.2.2.1 (c).

b. In Subsection 3.6.2.2.1 (e) provide a discussion (including references) of the methodology used to reduce the thrust coefficient factors of 1.26 and 2.0 by accounting for friction.

210.25 Subsection 3.6.2.3.3 states that piping integrity does not depend on pipe whip restraints for any piping design loading combination including earthquake. Subsection 3.2.1 states that pipe whip restraints need not remain functional in the event of a Safe Shutdown Earthquake. The staff agrees that pipe whip restraints do not have to be classified as Seismic Category 1, however, they should be designed to remain functional during a seismic event. Provide assurance that pipe whip restraints and their supporting structure cannot fail during a seismic event. If Subsection 3.8.3.3.2 is applicable to pipe whip restraints as well as their supporting structures, provide a reference to this Subsection in Subsection 3.6.2.3.3. Revise Subsections 3.2.1 and 3.6.2.3.3 to be consistent with the response to this question.

210.26 In Subsections 3.7.2.1.3, 3.7.3.3.1.3, and 3.7.3.8.2.1, the multiple support excitation analysis method is referenced as an alternative to the envelope response spectrum method when calculating inertial responses of multiply-supported piping and equipment. This alternate method is acceptable to the staff only under the following conditions:

a. The multiple support input response spectrum method may be used only when support group responses are combined by the absolute sum method.

b. The multiple support input response spectrum method may not be used in analyses which also use the damping values from ASME Code Case N-411, "Alternate Damping Valves for Seismic Analysis of Classes 1, 2, and 3 Piping Sections, Section III, Division 1." This position is one of the conditions listed in Regulatory Guide 1.84, Revision 24 for using Code Case N-411.

Provide a commitment to the above conditions in an appropriate Section in the SSAR and cross reference this commitment in Subsection 3.7.2.1.3, 3.7.3.3.1.3, 3.7.3.8.2.1 and any other subsection which discusses the multiple support exitation analysis alternative.

210.27 The information in Subsection 3.7.3.4, "Basis of Selection of Frequencies" does not appear to be consistent with the guidelines in Standard Review Plan, Section 3.9.2, Paragraph II.2.C. Revise Subsection 3.7.3.4 to include a commitment that, to avoid resonance, the fundamental frequencies of components and equipment should be selected to be less than 1/2 or more than twice the dominant frequencies of the support structure.

210.28 In Subsection 3.7.3.10, the statement is made that the vertical ground design response spectrum is used for equipment vertical seismic load determination if it can be shown that the structures supporting the equipment are rigid or quasi-rigid in the vertical direction. Provide definitions of "rigid, "quasirigid" and "support structure" in Sub-section 3.7.3.10.

210.29 Subsection 3.9.2.2.2.1 states that preliminary dynamic tests are conducted to verify the operability of the control rod drive (CRD) during a dynamic event. Provide a more detailed description of these tests and, if applicable, discuss how the results of the tests are correlated with the analysis of the CRD housing (with the enclosed CRD) which is mentioned in the first sentence of this subsection. If the fine motion control rod drive system is not included in these tests, describe how that system is seismically qualified.

210.30 Revise the discussion Subsection 3.9.1.4.4 to be consistent with the information in Subsection 3.9.3.4.3 for the reactor pressure vessel stabilizer and Subsection 3.9.3.5 for the supports for the fine motion control rod drive and in-core housings.

210.31 In Subsection 3.9.2.1.1, ANSI/ASME OM3-1987, "Requirements for Preoperational and Initial Startup Vibration Testing of Nuclear Power Plant Piping Systems" is referenced for vibration testing of ABWR piping systems. However, in Subsections 3.9.2.1.2 and 14.2.12.1, there is no reference to OM3 for preoperational thermal expansion and dynamic testing and the information in these subsections on these phases of preoperational testing is not presented in sufficient detail for the staff to evaluate. Revise Subsections 3.9.2.1.2 and 14.2.12.1 to either include a reference to ANSI/ASME OM3-1987 or present information similar to that for the Main Steam Line piping which is discussed in Subsections 3.9.2.1.3, 3.9.2.1.4, 3.9.2.1.5 and 3.9.2.1.6.

210.32 In Subsections 3.9.2.1.1 and 14.2.12.1, there is no mention of preoperational vibration testing of safety-related instrumentation lines. It is the staff's position that all essential safety-related instrumentation lines and small bore piping should be included in the vibration monitoring program during preoperational or start-up testing. We require that either a visual or instrumented inspection (as appropriate) be conducted to identify any excessive vibration that could result in fatigue failure. Generally, this includes the portion up to and including the first support away from the connection to large bore piping or component. If observations suggest that other spans are being excited, further inspection would be conducted on a case by case basis. Revise the above Subsections to provide a commitment to this position.

210.33 The discussions in Subsection 3.9.2.5 and 3.9.5.2 relative to the dynamic system analysis of reactor internals under faulted conditions does not provide enough detailed information for the staff to evaluate. Standard Review Plan, Section 3.9.2.II.5 provides the acceptance criteria which the staff uses to evaluate this issue. Information in sufficient detail to implement this criteria is required before the staff can complete its evaluation. Revise Subsection 3.9.2.5 to include this information either in the form of references or an additional appendix in Section 3-2 of the ABWR SSAR.

210.34 In Table 3.9-2, the acceptance criteria for the stresses resulting from the service loading combination of normal loads plus the most limiting safety/relief valve loads plus turbine stop valve closure induced loads is identified as ASME Level D Service Limits. If this is a typographical error, replace Level D with Level B in this table. If it is not an error, provide the justification for using Level D Service Limits for this loading combination.

210.35 Provide the basis for assuring that the feedwater isolation check valves can perform its intended function and satisfy GDC 54 and 55 following a feedwater line break outside containment. Additionally, discuss what actions have been taken to preclude the possibility of a feedwater pump trip transient causing a feedwater line break outside containment.

210.36 The discussions of ASME Class 1, 2 and 3 safety-related code components in Subsections 3.9.3.1.3 through 3.9.3.1.7 and 3.9.3.1.9 through 3.9.3.1.19 use the terms "designed and evaluated" in accordance with ASME Section III rules for Class 1, 2 and 3 components. In discussions of this nature, the word "constructed" should be used rather than "designed and evaluated" where construction is defined in accordance with the ASME Section III, Subsection NCA 1100 definition, i.e., "an all inclusive term comprising materials, design, fabrication, examination, testing, inspection and certification required in the manufacture and installation of items". Revise all of the above Subsections to state that all of these components are constructed in accordance with the ASME III NCA 1100 definition.

210.37 Subsection 3.9.3.2 contains several references to IEEE-344, "IEEE Recommended Practices for Seismic Qualification of Class IE Equipment for Nuclear Power Generating Stations" with no issue date.

To be consistent with current staff positions on this issue, revise each of these references to read "IEEE STD. 344-1987" and add a commitment to NRC Regulatory Guide 1.100, Revision 2, "Seismic Qualification of Electrical Equipment in Nuclear Power Plants" to each reference. The staff considers these two documents to be applicable to mechanical as well as electrical equipment.

210.38 Subsection 3.9.3.3.2, "Other Safety/Relief Valves" references ASME Section III, Appendix O for the safety-relief valve opening and pipe reaction loads which will be used in the design of ABWR safety-relief valves. The staff's position on this issue is that if Appendix O is used, the additional criteria in Standard Review Plan, Section 3.9.3, Paragraph II. 2 is applicable. Revise Subsection 3.9.3.3.2 to include a commitment to this position.

210.39 Subsections 3.9.3.4.1 and 3.9.3.5 both state that the jurisdictional boundary between component supports designed to ASME Section III, Subsection NF and the building structive shall be as defined in the project design specifications. The project design specifications may or may not agree with the definitions of jurisdictional boundaries which are in ASME Subsection NF. Therefore, revise Subsections 3.9.3.4.1

and 3.9.3.5 of the ABWR SSAR to provide a commitment that the 1987 Addenda to the 1986 Edition of ASME Section III, Subsection NF will be used to define the jurisdictional boundary between Subsection NF component supports and the building structure.

210.40 The information in Subsections 3.9.3.4.2 and 3.9.3.5 relative to analyses for buckling of the reactor pressure vessel support skirt and other ASME III component supports needs to be updated and clarified as follows:

a. Paragraph 1370 (c) of ASME III, Appendix F, which is referenced in both of the above subsections was deleted in the Summer, 1983 Addenda to ASME III, Division 1 Appendices. ASME Appendix XVII, which is referenced in Subsection 3.9.3.5 was deleted in the Winter, 1985 Addenda. Revise Subsections 3.9.3.4.2 and 3.9.3.5 to provide references which are applicable to the latest edition of ASME, Section III.

b. Provide a more detailed description of how the critical buckling strength of the RPV support skirt and other ASME III component supports will be determined.

210.41 The following additional information is required in Subsection 3.9.3.4 relative to the design of bolts for component supports:

1. Provide the allowable stress limits and/or safety factors which are applicable to bolts used in equipment anchorage, component supports and flanged connections.

   Specifically provide a discussion of the design methods applicable to expansion anchor bolts and case-in-place used in component supports and equipment anchorage.

210.42 In Subsection 3.9.3, provide the design basis which will be used in the ABWR to insure the structural integrity of safety-related heating, ventilation and air conditioning ductwork and its supports.

210.43 Subsection 3.9.4 outlines seven types of tests which will be used as a basis for the ABWR Control Rod Drive (CRD) Performance Assurance Program. The first type, "Development Tests" are discussed in Subsection 4.6.3.1. According to this discussion, at least three different prototype designs of the Fine Motion Control Rod Drive (FMCRD) have been subjected to various test programs. The staff's Question 440.8 requested the results of the tests of the inplant FMCRD prototype which are currently being conducted at La Salle, Unit 2. In addition to a response to Question 440.8, provide a description of the differences between the initial, inplant and reference FMCRD designs and, if applicable, a discussion of any correlation that may exist between the accumulated test data from all three designs and the design criteria discussed in Subsections 3.9.1.1, 3.9.1.4 and 3.9.3 and Table 3.9-2.

210.44 Subsection 3.9.5.1.1 states that the core support structures in the ABWR are classified as Safety Class 3. The staff's position is that these structures are necessary to help maintain core geometry and should therefore be classified as Safety Class 2 to obtain a higher level of quality assurance than Safety Class 3. Revise Tables 3.2-1 and 3.2.3 and Subsection 3.9.5.1.1 to agree with this position.

210.45 In Subsections 3.9.5.1.2.4, 3.9.5.1.2.5 and 3.9.5.1.2.6, the feedwater spargers, RHR/ECCS low pressure flooder spargers and the ECCS high pressure core flooder spargers and piping are all classified as Safety Class 3. The staff's position is that these reactor internal components are necessary to help accomplish the safety function of emergency core cooling and should therefore be classified as Safety Class 2 to obtain a higher level of quality assurance than Safety Class 3. Revise Table 3.2-1 and Subsections 3.9.5.1.2.4, 3.9.5.1.2.5 and 3.9.5.1.2.6 to agree with this position.

210.46 Portions of the stress, deformation and buckling limits for safety class reactor internals which are listed in Tables 3.9-4, 3.9-5 and 3.9-6 requires additional review by the staff. If either Equation b in Table 3.9-4, Equations e, f, and g in Table 3.9-5 or Equation c in Table 3.9-6 will be used in the design of safety class reactor internals for the ABWR, provide a commitment in each of these tables that supporting data will be provided to the staff for review.

210.47 The information in Subsection 3.9.6 infers that only ASME Class 1, 2 and 3 pumps and valves will be included in the inservice testing (IST) program for the ABWR. It is the staff's position as stated in Standard Review Plan, Sections 3.9.6.II.1 and 3.9.6.II.2 that all pumps and valves which are considered as safety-related should be included in the IST program even if they are not categorized as ASME Class 1, 2 or 3. Revise Subsection 3.9.6 to agree with this position.

210.48 The first paragraph in Subsection 3.9.6 states that accessibility for inservice testing of applicable pumps and valves is provided in the plant design. However, the second paragraph and Subsection 3.9.6.3 infers that relief from ASME Section XI inservice testing will be submitted for some pumps and valves.

All of the plants which have been licensed by NRC so far have been allowed to request relief from the ASME Section XI inservice testing rules for a limited number of pumps and valves. These pumps and valves are generally installed in systems in which it is impractical to meet the Section XI rules because of limitations in the system design which make the pump or valve difficult to test without additional design changes. Therefore, the staff granted many of these requests for relief because imposition of these rules would have resulted in hardships to the licensee without a compensating increase in the level of safety. The underlying reason for the regulation allowing these reliefs from the code was that the detailed piping system designs for all of these plants was completed prior to the time that the staff began to implement the ASME Section XI rules.

A plant such as the ABWR, for which the final design is not complete, has sufficient lead time available to include provisions for this type of testing in the detailed design of applicable piping systems. Therefore, requests for relief from the applicable ASME Section XI testing rules for pumps and valves will not be granted for the ABWR. Revise Subsection 3.9.6 to provide a more explicit commitment that ABWR piping systems will be designed to accommodate the applicable code requirements for inservice testing of pumps and valves. However, with regard to subsequent or future code revisions to the applicable ASME Code for the ABWR plants, requests for relief from certain updated code requirements may still be submitted for staff review in accordance with 10 CFR 50.55a(g).

210.49 In Subsection 3.9.6, "Inserivce Testing of Pumps and Valves," provide a commitment to perform periodic leak testing of all pressure isolation valves in accordance with the applicable sections of the Technical Specifications for recently licensed BWR/6 plants. Normally, this information includes a list of all pressure isolation valves which will be leak tested. If such a list is not available for the ABWR, a commitment to provide the list of valves as a part of the ABWR Technical Specifications will be acceptable.

210.50 In accordance with NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," the staff is currently requesting licensees and applicants to review systems connected to the reactor coolant system to determine whether any sections of such piping which cannot be isolated can be subjected to stresses from temperature stratification or temperature oscillations that could be induced by leaking valves. If this phenomenon was not considered in the design analysis of the ABWR piping, submit a response to action Item 3 in Bulletin 88-08 which will be applicable.

Seismic and dynamic load qualification

271.01 Subsection 3.10.1.3 states that the ABWR program for dynamic qualification of Seismic Category 1 electrical equipment meets the criteria contained in IEEE-344 as modified and endorsed by Regulatory Guide 1.100. To be consistent with recent staff positions on this issue, revise Subsection 3.10.1.3 [illegible] "[illegible]E-344-1987 as modified and endorsed by Regulato[illegible]de 1.100, Revision 2".

271.02 Subsection 3.10.1.3, "Dynamic Qualification Program" states that Section 4.4 of GE's Environmental Qualification Program (NEDE-24326-1-P) will be used for dynamic qualification of Seismic Category 1 electrical equipment and that this report is referenced in Subsection 3.11. The reference in Subsection 3.11.7 is to the January, 1983 version of NEDE-24326-1-P. The staff's approval of this report is based on the January, 1986 Revision. Revise Reference 2 in Subsection 3.11.7 to change the date of NEDE-24326 from January, 1983 to January, 1986.