Docket No. STN 50-605

November 21, 1989

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 Evaluations (PDSERs) relating to the staff's review of your application for certification of the Advanced Boiling Water Reactor Design. In these PDSERs we have identified a need for additional information in the form of outstanding issues. Enclosure 1 was prepared by the Mechanical Engineering Branch and Enclosure 2 was prepared by the Structural and Geosciences Branch. The staff plans to discuss Enclosure 2 with GE during the seismic design audit scheduled for November 28-30, 1989 in San Jose. Enclosure 2 will be discussed at a later date. However, 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 January 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:
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ENCLOSURE 1

PRELIMINARY DRAFT SAFETY EVALUATION INPUT FOR
THE ADVANCED BWR STANDARD SAFETY ANALYSIS REPORT,
DOCKET NO.: 50-605
MECHANICAL ENGINEERING BRANCH

3. DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.1 General

Section 3.1 of the ABWR SSAR discusses conformance of structures, components, equipment, and systems to the General Design Criteria (GDC) in 10 CFR 50, Appendix A. Using this information, the staff has reviewed the design criteria to verify that the ABWR Nuclear Island will be designed to meet the GDC.

The staff review of structures, components, equipment, and systems relies heavily on the application of industry codes and standards that have been used as accepted industry practice. The codes and standards cited in this report have been previously reviewed by the staff, found acceptable, and incorporated into the SRP (NUREG-0800).

3.2 Classification of Structures, Systems, and Components

3.2.1 Seismic Classification

GDC 2, "Design Bases for Protection Against Natural Phenomena," in part, requires that nuclear power plant structures, systems, and components important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety function. Certain of these features are necessary to ensure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to 10 CFR 100 guideline exposures. The earthquake for which these safety-related plant features are designed is defined as the safe shutdown earthquake (SSE) in 10 CFR 100, Appendix A. The SSE is based on an evaluation of the maximum earthquake potential and is that earthquake which produces the maximum vibratory ground motion for which structures, systems, and components are designed to remain functional. Those plant features that are designed to remain functional if an SSE occurs are designated seismic Category I in Regulatory Guide (RG) 1.29, "Seismic Design Classification." The ABWR SSAR was reviewed in accordance with SRP Section 3.2.1 (NUREG-0800) which references RG 1.29.

The structures, systems, components, and equipment of the ABWR Nuclear Island that are required to be designed to withstand the effects of an SSE and remain functional have been identified in Table 3.2-1 of the SSAR. This table, in part, identifies major components in fluid systems, mechanical systems, and associated structures designated as seismic Category I. The staff has reviewed Table 3.2-1 and other applicable information in the SSAR and concluded that the structures, systems and components important to safety in the ABWR Nuclear Island have been properly classified as Seismic Category I items in conformance with RG 1.29. All other structures, systems, and components that may be required for operation of the facility are not required to be designed to seismic Category I requirements, including those portions of Category I systems such as vent lines, fill lines, drain lines, and test lines on the downstream side of isolation valves and portions of these systems that are not required to perform a safety function.

Note f in Table 3.2-1 and Subsections 3.7.2.8 and 3.7.3.13 in the SSAR state that equipment, structures and piping in the ABWR that is non-seismic Category I but which could damage Seismic Category I items if its structural integrity failed is analyzed and designed to assure its integrity under seismic loading from the Safe Shutdown Earthquake. At the interface between Seismic and non-seismic Category 1 piping systems, the Seismic Category 1 dynamic analysis will be extended to either the first anchor point in the non-seismic system or to sufficient distance in the non-seismic system as so as not to degrade the accuracy of the Seismic Category 1 analysis. The staff has concluded that this commitment is in conformance with R.G. 1.29.

The staff has concluded that the above information constitutes a basis for satisfying applicable portions of GDC 2 in Appendix A of 10 CFR 50, and is therefore acceptable.

3.2.2 System Quality Group Classification

GDC 1, "Quality Standards and Records," in 10 CFR 50, Appendix A requires that nuclear power plant structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Important to safety is defined in the introduction to 10 CFR 50, Appendix A as "structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public ". A subset of important to safety is the term safety related (safety grade). 10 CFR 50.49(b)(1), and 10 CFR 100, Appendix A, Sections III(c), VI(a)(1) and VI(b)(3) defines safety related as those systems, structures and components necessary to assure either:

(1) The integrity of the reactor coolant pressure boundary,
(2) The capability to shut down the reactor and maintain it in a safe shutdown condition, or
(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the 10 CFR Part 100 guidelines.

In the ABWR SSAR, 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. This ABWR position, which is restated in the applicants' response to the staffs' question 210.3, is not consistent with the above definitions and is not acceptable. A strict interpretation of the ABWR position could result in unacceptable quality group classifications of certain structures, systems and components in the ABWR SSAR Table 3.2-1, "Classification Summary". As discussed in NRC Generic Letter 84-01, "NRC Use of the Terms Important to Safety and Safety Related", dated January 5, 1984, the staff's evaluations of the quality assurance requirements in 10 CFR Part 50, Appendix B have generally applied the narrower class of "safety-related" equipment as defined above. 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. These specific situations have occurred during the staff's reviews of systems quality group classifications in the Safety Analysis Reports for previous BWR and PWRs and have generally resulted in the imposition of additional quality assurance commitments from the applicants commensurate with the importance to safety of the equipment involved.

With the above position from G.L. 84-01 as a guideline, the staff reviewed the ABWR SSAR in accordance with Standard Review Plan, Section 3.2.2 (NUREG-0800), "System Quality Group Classification". SRP 3.2.2 references Regulatory Guide 1.26, "Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," as the principal document used in the staff review for identifying on a functional basis the pressure retaining components of those systems important to safety as NRC Quality Groups A, B, C, or D. 10 CFR 50.55a identifies those American Society of Mechanical Engineers Boiler and Pressure Vessel Code (AMSE) Section III, Class 1 components that are part of the RCPB. Conformance of these RCPB components with 10 CFR 50.55a is discussed in Section 5.2.1.1 of this SER. These RCPB components are designated in RG 1.26 as Quality Group A. Certain other RCPB components that meet the exclusion requirements of 10 CFR 50.55a(c)(2) are classified Quality Group B in accordance with RG 1.26.

The applicant used American Nuclear Society (ANS) Safety Classes 1, 2, 3, and Non-Nuclear Safety (NNS) as defined in American National Standard ANSI/ANS 52.1-1983, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants", in the classification of system components as an alternative method of meeting the guidance of RG 1.26. Tables 3.2-2 and 3.2-3 in the SSAR provide a correlation between (1) ABWR Safety Classes 1, 2, 3, and NNS, (2) the Commission's Quality Groups A, B, C, and D in RG 1.26 and (3) ASME Section III

Code Classes. A summary of the relationship between the three methods of classification in the SSAR is listed in the table below. It should be noted that the following table is only applicable to pressure-retaining components.

NRC Quality Group	ABWR Safety Class	ASME Section III Code Class
A	1	1
B	2	2
C	3	3
D	NNS	-

The above table is acceptable for defining the relationship between the three methods of classification for pressure retaining components. However, as stated in Question 210.4, the staff has not endorsed ANSI/ANS 52.1-1983 and cannot rely on the safety classifications in this document in determining the acceptability of non-pressure retaining structures, systems and components. Therefore, for non-pressure retaining components, the staffs' review of Table 3.2-1 in the SSAR has concentrated on an evaluation of Quality Assurance in accordance with 10 CFR 50, Appendix B and seismic classifications. Listed below is a brief summary of each unresolved issue relative to Table 3.2-1. Each issue is identified by the applicant's responses to staff questions which appear in SSAR Subsection 20.30.5, Amendment 3. The Item numbers are those listed in Table 3.2-1.

_ RESPONSE to Q 210.6
The staff cannot complete its review of the main steam line quality group and seismic classifications until the issue of the Main Steam Line Isolation Valve Leakage Control System (MSIV-LCS) is resolved. Resolution of this issue, which is discussed in Section 6.7 of this Draft SER, may result in revisions of Table 3.2-1, Figure 5.1-3b, Subsection 3.9.3.1.3 and Subsection 5.4.9.3.

_ RESPONSE to Q 210.7
The response to this question in Amendment 3 to the SSAR was acceptable except that Item B2.7, "Feedwater piping from shutoff valve to seismic interface restraint" was added to Table 3.2-1 with a Quality Group D classification. This classification is not consistent with the remainder of the response to this question and is unacceptable. Item B2.7 should be deleted from the table.

_ RESPONSE TO Q 210.8
The response to this question stated that the primary side of the recirculation motor cooling system is designed to ASME Section III Class 1 criteria but is classified as Safety Class 3 and Quality Group C. As stated in Question 210.8, the staff does not agree with this classification. The staff considers this system to be connected to the reactor coolant pressure boundary and therefore should be considered a part of that boundary in accordance with 10 CFR Part 50.2. To be consistent with this position and the information

in Subsection 3.9.3.1.4 and Figure 5.4-4 in the SSAR, the applicant is requested to change the classifications of Items B3.1, "Piping - primary side, motor cooling system" and B3.2, "Pipe Supports" to Safety Class 1 and Quality Group A.

_ RESPONSE TO Q 210.11
Part of the applicant's response to this question stated that the containment spray piping within the outermost isolation valve is Safety Class 2 because it is not a part of the reactor coolant pressure boundary. Since this is equivalent to Quality Group B, the staff agrees with this classification. However, Figures 5.4-10a, "RHR System P&ID" and 6.2-38a, "Plant Requirements, Group Classification and Containment Isolation Diagram" in the SSAR both show a Quality Group transition point from Quality Group B to C in this piping at the containment boundary. This transition point should be deleted in these figures and the containment spray piping inside containment should remain Quality Group B and Safety Class 2.

Another part of the applicants' response to this question stated that the containment spray spargers are Safety Class 3 because they are not part of either the reactor coolant pressure boundary or the pressure boundary of an engineered safety feature piping system. The staff's position is that since these spargers are part of a Safety Class 2 piping system as discussed in the above paragraph, they should also be Safety Class 2. This part of the response to Q 210.11 should be revised, and in Table 3.2-1 of the SSAR, Item E1.3 should either be revised or an additional item added in E1 of this table to address the containment spray piping and spargers.

_ RESPONSES TO Q 210.11 AND Q 210.45
The staff position is that reactor internals such as the feedwater spargers, RHR/ECCS low pressure flooder spargers and ECCS high pressure core flooder spargers are necessary to help accomplish the safety function of emergency core cooling and should therefore be classified as Safety Class 2 and Quality Group B to obtain a higher level of quality assurance than Safety Class 3. The applicants response to Q 210.45 stated that these spargers cannot be pressure tested and therefore cannot meet the requirements of Safety Class 2. The system hydrostatic test requirements for ASME Class 2 and Class 3 systems are almost identical in ASME Section III, Subsections NC and ND. In addition, NC and ND 6114.1(b) "System Pressure Test" for Class 2 and 3 systems respectively, provides an option for testing open ended systems and states that spray nozzles and their attachment weld or mechanical joints need not be tested. The following parts of the SSAR should be revised to reflect this staff position:

1. Table 3.2-1, Items B1.5 E1.3 and E4.1
2. Subsections 3.9.5.1.2.4, 3.9.5.1.2.5 and 3.9.5.1.2.6
3. Responses to Q 210.11 and Q 210.45

RESPONSE to Q 210.12
Figures 6.3-7a and 6.3-7b, "High Pressure Core Flooder System P&ID" in the SSAR appears to classify all piping within the outermost isolation valves (F004B and F014.01C) in this system as Quality Group A and Safety Class 1. This does not appear to agree with the response to this question which states that portions of this system which are part of an engineered safety feature and are within the outermost isolation valve, but are not part of the reactor coolant pressure boundary (RCPB) are Safety Class 2. The applicant is requested to revise the response to Q 210.12 by identifying all portions of the High pressure Core Flooder System piping which is within the outermost isolation valve and is Safety Class 2 or 3. Items E2.1 and E2.5 in Table 3.2-1 of the SSAR should also be revised to identify Class 2 and 3 piping and valves which are within the outermost isolation valve.

RESPONSE TO Q 210.13
The response to this question references ANSI/ANS 57.1-1980, "Design Requirements for LWR Fuel Handling Systems" as the basis for the Non-Nuclear Safety classification of Item F4.1 "Refueling Equipment Platform Assembly" in Table 3.2-1 of the SSAR. The staff position is that the Refueling Equipment Platform Assembly is important to safety and, as a minimum should meet applicable quality assurance requirements of Appendix B to 10 CFR 50 in addition to being classified as Seismic Category 1. This position on Appendix B is consistent with the guidelines in Regulatory Guide 1.29. Item F4.1 in Table 3.2-1 should be revised to reflect this position.

RESPONSE TO Q 210.15
The response to this question references the ANSI/ANS 52.1 - 1983 standard as the basis for the Non-Nuclear Safety classification of the new and spent fuel storage racks and the defective fuel storage container. The staff position is that the new and spent fuel storage racks and the defective fuel storage container are important to safety and, as a minimum, should meet applicable quality assurance requirements of Appendix B to 10 CFR 50 in addition to being classified as Seismic Category 1. This position on Appendix B is consistent with R.G.1.29. Items F5.1 and F5.2 in Table 3.2-1 of the SSAR should be revised to reflect this position.

RESPONSE TO Q 210.19
The response to this question is not acceptable. Reference the discussions above on the responses to Q 210.13 and 210.15

Subsequent to resolutions of the issues discussed above relative to Table 3.2-1 in the SSAR, the staff will submit its evaluation in the Final SER.

3.6.2 Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

General Design Criterion 4, "Environmental and Missile Design Bases," of 10 CFR Part 50, Appendix A, requires that structures, systems, and components important to safety shall be designed to be compatible with and to accommodate the effects of the environmental conditions as a result of normal operations, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be adequately protected against dynamic effects (including the effects of missiles, pipe whipping, and discharging fluids) that may result from equipment failures and from events and conditions outside the nuclear power plant.

The staff's review, conducted in accordance with Standard Review Plan (NUREG-0800), Section 3.6.2, "Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," pertains to the methodology used for protecting safety-related structures, systems, and components against the effects of postulated pipe breaks both inside and outside containment. The staff has used the review procedures identified in SRP 3.6.2 to evaluate the criteria and methodology used by the applicant to determine the effect that breaks in high energy fluid systems would have on adjacent safety-related structures, systems, or components with respect to jet impingement and pipe whip. The details of the staff's review follow.

Pipe whip need only be considered in those high-energy piping systems having fluid reservoirs with sufficient capacity to develop a jet stream. The criteria for determining high- and moderate-energy lines is found in Branch Technical Position ASB 3-1 of Standard Review Plan 3.6.1, "Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment." These criteria have been correctly used by the applicant in Subsection 3.6.2.1 of the SSAR. A list of all high energy systems is included in Tables 3.6-3 and 3.6.4 of the SSAR.

In the ABWR Nuclear Island, breaks are not postulated in those portions of high energy piping between the containment isolation valves outside and inside containment which are designed to meet the requirements of ASME Code, Section III, Subarticle NE-1120, and the additional design guidelines in NUREG-0800 SRP Section 3.6.2, including Branch Technical Position MEB 3-1, Revision 2 dated June 1987. A part of these guidelines recommends that an augmented inservice inspection program be implemented for those portions of piping within the break exclusion region. The applicant has committed to perform a 100% volumetric examination of circumferential and longitudinal pipe welds in the break exclusion region during each inspection interval as defined in IWA-2400, ASME Code, Section XI. The staff finds that the above commitment for the design and examination of high energy piping in the containment penetration area meets the guidelines in SRP 3.6.2 and is acceptable.

For ASME Class 1, 2 and 3 and non-ASME Seismic Category 1 high and moderate energy lines which are not in the containment penetration area, the applicant, in Subsection 3.6.2 of the SSAR has presented criteria used for determination of postulated rupture and crack locations and methodology used to evaluated the dynamic effects of pipe whip, jet thrust and jet impingement which result from such breaks. The applicants criteria meets the guidelines of SRP 3.6.2 and is therefore acceptable.

The SRP 3.6.2 also requests that for the FDA, the applicant should include the following in the SSAR:

- Sketches of applicable piping systems showing the location, size and orientation of postulated pipe breaks and the location of pipe whip restraints and jet impingement barriers.

- A summary of the data developed to select postulated break locations. This should include calculated stress intensities, cumulative usage factors and stress ranges as delineated in SRP 3.6.2, MEB 3-1.

The applicant is requested to incorporate the above information in the SSAR.

The response to the staff's Question 210.25 requires an editorial change in Subsection 3.6.2.3.3 of the SSAR to state that pipe whip restraints shall remain functional following an earthquake up to and including an SSE.

Subsequent to the applicants submittal of the information requested above relative to pipe break locations and stress calculations, the staff's conclusion in its final SER will be as follows:

On the basis of its review of Section 3.6.2 in the SSAR, the staff concludes that the criteria for postulating pipe rupture locations and the methodology for evaluating the subsequent dynamic effects of these ruptures are in accordance with SRP 3.6.2, meet GDC 4 and, therefore, are acceptable. This conclusion provides the following assurances:

1. The proposed pipe rupture locations have been adequately assumed and the design of piping restraints and measures to deal with the subsequent dynamic effects of pipe whip and jet impingment provide adequate protection to the structural integrity of safety-related structures, systems and components.

2. The provision for protection against dynamic effects associated with pipe ruptures of the reactor coolant pressure boundary inside containment and the resulting discharging fluid will provide adequate assurance that design-basis loss-of-coolant

accidents will not be aggravated by the sequential failures of safety-related piping and emergency core cooling system performance will not be degraded by these dynamic effects.

3. The proposed piping and restraint arrangement and applicable design considerations for high and moderate-energy fluid systems inside and outside containment, including the reactor coolant pressure boundary, will provide adequate assurance that the structures, systems, and components important to safety that are in close proximity to the postulated pipe ruptures will be protected. The design is of a nature to mitigate the consequences of pipe ruptures so that the reactor can be safely shut down and maintained in a safe shutdown condition in the event of a postulated rupture of a high or moderate-energy piping system inside or outside containment.

3.9 Mechanical Systems and Components

The review performed under SRP Sections 3.9.1 through 3.9.6 (NUREG-0800) pertains to the structural integrity and functional capability of various safety-related mechanical components in the plant. The staff's review is not limited to ASME Code components and supports, but is extended to other components such as control rod drive mechanisms, certain reactor internals, and any safety-related piping designed to industry standards other than the ASME Code. The staff reviews such issues as load combinations, allowable stresses, methods of analysis, summary of results, and preoperational testing. The staff's review must arrive at the conclusion that there is adequate assurance of a mechanical component performing its safety-related function under all postulated combinations of normal operating conditions, system operating transients, postulated pipe breaks, and seismic events.

3.9.1 Special Topics for Mechanical Components

The staff has reviewed the information in Subsection 3.9.1 of the SSAR relative to the design transients and methods of analysis used for all seismic Category 1 components, component supports, core support structures, and reactor internals designated as Class 1, 2, 3 and CS under ASME Code Section III, and those not covered by the Code. The assumptions and procedures used for the inclusion of transients in the design and fatigue evaluation of ASME Code Class 1 and CS components have been reviewed. The staff's review also covered the computer programs used in the design and analysis of seismic Category 1 components and their supports, as well as experimental and inelastic analytical techniques.

In Table 3.9.1 of the SSAR, the applicant has provided a list of the design transients for five plant operating conditions and the number of either plant operating events or cycles for each of the design transients which will be used in the design and fatigue analyses of the reactor pressure vessel. The operating conditions included the following:

1. ASME Service Level A - Normal conditions.

2. ASME Service Level B - Upset conditions - Incidents of moderate frequency.

3. ASME Service Level C - Emergency conditions - Infrequent incidents.

4. ASME Service Level D - Faulted conditions - Low probability postulated events.

5. Testing conditions

The number of events or cycles resulting from each of the listed design transients which are applicable to other ASME Class 1 components are documented in the design specification and/or stress report for each component.

The applicant used computer codes to analyze mechanical components. A description of all computer programs used by the applicant for static and dynamic analyses to determine the structural and functional integrity of Seismic Category I Code and non-Code items is included in Appendix 3D of the SSAR. Design control measures to verify the adequacy of the design of safety-related components are required by 10 CFR 50, Appendix B. In Subsection 3.9.1.2 of the SSAR, a commitment is made that the quality of the programs and the computer results are controlled either by General Electric or by outside computer program developers. In addition, the programs are verified by one or more of the methods recommended in SRP 3.9.1.

In Subsection 3.9.1.3 of the SSAR, the applicant identified several components for which experimental stress analysis is performed in conjunction with analytical evaluation. The experimental stress analysis methods are used in compliance with Appendix II of the ASME Code, Section III. This commitment meets the guidelines in SRP Section 3.9.1.

The applicant has not identified any components which are evaluated by inelastic analysis methods. The applicant is requested to either submit a statement in the SSAR that inelastic methods are not used, or identify any applicable components and submit information relative to the inelastic analyses which meets the guidelines of SRP 3.9.1.

When acceptable information on inelastic analyses has been received the staff's evaluation for Section 3.9.1 of the Drafty SER will be as follows:

On the basis of its review of Subsection 3.9.1 of the SSAR, the staff concludes that the design transients and resulting load combinations with appropriate specific design and service limits for mechanical components and supports are acceptable and meet the applicable portions of GDC 1, 2, 14, and 15; 10 CFR 50, Appendix B, 10 CFR 50, Appendix A and Standard Review Plan, Section 3.9.1.

The applicant has met GDC 14 and 15 by demonstrating that the design transients and resulting loads and load combinations with appropriate specific design and service limits that the applicant has used for designing ASME Code Class 1 and CS components and supports and reactor internals provide a complete basis for design of the reactor coolant pressure boundary for all conditions and events expected over the service lifetime of the plant.

The applicant has met the relevant requirements of GDC-2 and 10 CFR 100, Appendix A by including seismic events in design transients which serve as design basis to withstand the effects of natural phenomena.

The applicant has met 10 CFR 50, Appendix B, and GDC 1 by having submitted information that demonstrates the applicability and validity of the design methods and computer programs used for the design and analysis of seismic Category 1 ASME Code Class 1, 2, 3, and CS structures and non-code structures within the present state-of-the-art limits and by having design control measures that are acceptable to ensure the quality of the computer programs.

3.9.2 Dynamic Testing and Analysis of Systems, Components and Equipment

The staff has reviewed the methodology, testing procedures and dynamic analyses employed by the applicant to ensure the structural integrity and functionality of piping systems, mechanical equipment, and their supports under vibratory loadings. The staff's review included (1) the piping vibration, thermal expansion, and dynamic effects testing; (2) the seismic system analysis methods; (3) the dynamic responses of structural components within the reactor caused by the steady-state and operational flow transient conditions; (4) flow-induced vibration testing of reactor internals to be conducted during the preoperational and startup test program; and (5) the dynamic analysis methods used to confirm the structural design adequacy and functional capability of the reactor internals and piping attached to the reactor vessel when subjected to loads from a loss-of-coolant accident (LOCA) in combination with a safe shutdown earthquake (SSE).

3.9.2.1 Piping Preoperational Vibration and Dynamic Effects Testing

Piping vibration, thermal expansion, and dynamic effects testing will be conducted during a preoperational testing program. The purpose of these tests is to ensure that the piping vibrations are within acceptable limits and that the piping system can expand thermally in a manner consistent with the design intent. During the plant's preoperational and startup testing program, the applicant will test various piping systems for abnormal, steady-state, or transient vibration and for restraint of thermal growth. Systems to be monitored will include (1) ASME Code Class 1, 2, and 3 piping systems; (2) high-energy piping systems inside seismic Category I structures; (3) high-energy portions of systems whose failure could reduce the functioning of seismic Category I plant features to an unacceptable safety level; and (4) seismic Category I portions of moderate-energy piping systems located outside containment. Steady-state vibration, whether flow induced or caused by nearby vibrating machinery, could cause 10_ or 10_ cycles of stress in the pipe during the 40 year life of the plant. For this reason, the staff requires that the stresses associated with steady-state vibration be minimized and limited to acceptable levels. The test program will consist of a mixture of instrumented measurements and visual observations by qualified personnel.

The information in Subsection 3.9.2.1 and 14.2.12 of the SSAR provides a general discussion of the proposed piping preoperational test program for the ABWR Nuclear Island. The staffs' current position on this issue is that for FDA, a specific commitment is required to develop a test program which will meet all of the rules in ANSI/OM-3, 1987, "Requirements for Preoperational and Initial Start-up Vibration Testing of Nuclear Power Plant Piping Systems" and ANSI/OM-7, "Requirements for Thermal Expansion Testing of Nuclear Power Plant Piping Systems." The staff finds that these criteria will provide an acceptable level of safety for a piping system to withstand the effects of vibration and thermal expansion during the plants' 40 year life. In Subsection 3.9.2.1 of the SSAR, the applicant has provided a general commitment to implement preoperational test programs based on ANSI/ASME OM-3, 1987, and ANSI/ASME OM-7, September 1986 (Draft Revision 7). The staff requests that the applicant provide a specific commitment in Subsection 3.9.2.1 that the test programs will meet all of the rules in these two standards. In addition, the staff requests a commitment that these same test programs will be conducted on all ABWR plants which will be constructed in accordance with the Design Certification.

Subsequent to receipt of the above commitment, the staffs evaluation will be as follows:

On the basis of its review of Subsections 3.9.2.1 and 14.2.12 of the SSAR, the staff concludes that the applicant will meet GDC 14 and 15 with respect to the design and testing of the reactor coolant pressure boundary. This provides reasonable assurance that rapidly propagating failure and gross rupture will not occur as a result of vibratory loadings. In addition, the testing ensures that design conditions will not be exceeded during normal operation, including anticipated operational occurrences, by having an acceptable vibration, thermal expansion, and dynamic effects test program that will be conducted during startup and initial operation of specified high and moderate energy piping, including all associated restraints and supports. The tests provide adequate assurance that the piping and piping supports will be designed to withstand vibrational dynamic effects as a result of valve closures, pump trips, and other operating modes associated with the design basis flow conditions. In addition, the tests provide assurance that adequate clearances and free movement of snubbers will exist for unrestrained thermal movement of piping and supports during normal system heatup and cooldown operations. The planned tests will develop loads similar to those experienced during transient and normal reactor operations.

3.9.2.2 Seismic Subsystem Analysis

The staffs review of this subject was performed using the guidelines of Standard Review Plan, Section 3.9.2 and consisted of an evaluation of Subsection 3.7.3 of the SSAR, "Seismic Subsystem Analysis". Areas reviewed were seismic analyses methods, determination of the number of earthquake cycles, basis for selection of frequencies, the combination of modal responses and spatial components of an earthquake, criteria used for damping, torsional effects of eccentric masses, interaction of other piping with seismic Category 1 piping, and Category 1 buried piping systems.

The system and subsystem analyses are performed by the applicant on an elastic basis. Modal response spectrum, multi-degree of freedom and time history methods form the basis for the analyses of all major seismic Category I systems and components. When the response spectrum method is used, modal responses are combined by the square-root-sum-of-the-squares (SRSS) rule. Closely spaced modes are combined using the criteria of Regulatory Guide 1.92. The applicant has considered all modes with frequencies below 33 hz in computing equipment and component response for seismic loadings.

For the dynamic analysis of seismic Category I piping, each system is idealized as a mathematical model consisting of lumped masses connected by elastic members. The stiffness matrix for the piping system is determined using the elastic properties of the pipe. This includes the effects of torsional, bending, shear, and axial deformations as well as change in stiffness due to curved members. Next, the mode shapes and the undamped natural frequencies are obtained. The dynamic response of the system is calculated by using the response spectrum method of analysis. For a piping system which is supported at points with different dynamic excitations, the response analysis is performed using an enveloped response spectrum. As an alternative to the enveloped response spectrum method, the applicant has chosen to use the multiple support excitation analysis method. This method is acceptable to the staff only if the support group responses are combined by the absolute sum method. In its response to the staff's Question 210.26, the applicant has maintained its position of combining these responses by the square-root-of-the-sum-of-the-squares (SRSS) method. The applicant has based its position on the results of various independent studies performed by industry and national laboratories in recent years. Certain issues concerning the applicability of the SRSS method have not been resolved to the staff's satisfaction. Therefore, the staff has not yet accepted the applicant's position. The applicant is requested to revise its response to Question 210.26 to conform to the staff position.

In response to the staff's Question 210.28, the applicant has stated that when a static analysis is performed in lieu of a dynamic analysis, a peak response multiplier of less than 1.5 may be used if justified. If a factor less than 1.5 is to be used, the staff requests the applicant to submit its justification in the SSAR for the staff's review.

Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," contains recommended values of damping to be used in the seismic analysis of structures systems, and components. In addition, Regulatory Guide 1.84, "Design and Fabrication Code Case Acceptability, ASME Section III, Division 1," Revision 25, May 1988, conditionally endorses ASME Code Case N-411-1, "Alternative Damping Values for Response Spectra Analysis of Classes 1, 2, and 3 Piping, Section III, Division 1". The damping values used by the applicant are the same as those specified in either Regulatory Guide 1.61 or ASME Code Case N-411-1 as permitted by Regulatory Guide 1.84. The staff finds these criteria to be acceptable.

In Subsection 3.7.3.12 of the SSAR, the applicant has outlined criteria which will be used in the analysis of buried Seismic Category 1 piping systems. These criteria conform with applicable guidelines in Standard Review Plan, Section 3.9.2 and are acceptable.

In the SSAR, Subsection 3.7.3.13, "Interaction of Other Piping with Seismic Category 1 Piping," the applicant provided criteria which conforms with applicable guidelines in SRP 3.9.2 and is acceptable. Part of this acceptable criteria states that when non-Seismic Category 1 piping cannot be isolated from Seismic Category 1 piping, the non-seismic piping is designed to withstand the Safe Shutdown Earthquake event to avoid jeopardizing the Category 1 piping. The implication is that the seismic event is assumed to cause a break in the non-seismic piping. However, in response to the staff's Question 210.23, the applicant revised Subsection 3.6.1.1.3(2) in the SSAR to state in part, that a seismic event does not initiate a pipe break event in non-Seismic Category 1 piping. The staff requests the applicant to revise the response to Question 210.23 to clarify this apparent discrepancy.

In the introduction to Section 3.7, "Seismic Design" in the SSAR, the applicant states that although this Section addresses seismic aspects of design and analysis, the methods of this Section are also applicable to other dynamic loads. However, Subsections 3.7.3.3.2, 3.7.3.4, 3.7.3.8.1.2, 3.7.3.8.1.4 and 3.7.3.10, all either state or infer that if the natural frequencies of equipment, components or subsystems are greater than 33 Hz, the item will be considered rigid and analyzed statically. This criterion appears to be applicable only to seismic loads and excludes input from suppression pool dynamic loads which could result in frequencies significantly greater than 33 Hz. The applicant is requested to either provide a basis for limiting all dynamic analyses of equipment, components or subsystems to those with natural frequencies below 33 Hz or revise applicable portions of Section 3.7 to include criteria which envelop seismic and other dynamic loads.

Subsequent to resolution of the issues applicable to Section 3.9.2.2 of the SER which are discussed above, the staff's evaluation of this section will be as follows:

On the basis of its review of Section 3.7.3 of the SSAR and the above information, the staff concludes that the applicant has met the relevant requirements of General Design Criterion 2 with respect to demonstrating the design adequacy of all Category I piping systems, components, and their supports to withstand earthquakes by meeting the regulatory positions of Regulatory Guides 1.61 and 1.92 or acceptable alternatives and by providing acceptable seismic analysis procedures and criteria which are consistent with applicable guidelines in SRP 3.9.2. The scope of review of the seismic subsystem analysis included the seismic analysis methods of all Cateogry I piping systems, components, and their supports. It included review of procedures for modeling, and inclusion of torsional effects, seismic analysis of multiply-supported equipment with distinct inputs, and determination of composite damping. The review has included design criteria and procedures for evaluation of buried piping and the interaction of non-Category I piping with Category I piping. The review has also included criteria and seismic analysis procedures for reactor internals.

3.9.2.3 Peoperational Flow-Induced Vibration Analysis and Testing of Reactor Internals

The configuration of reactor internals in the ABWR is different from the configuration in previous BWRs. Therefore, the applicant in Subsection 3.9.2.4 of the SSAR, has stated that the first ABWR plant will be considered a prototype and will be tested in accordance with the guidelines for prototype plants in Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing." These tests will be discussed in more detail below. Prior to this testing, the dynamic responses of reactor internals to steady-state conditions and operational flow transients must be predicted for prototype plants. One of the first steps involved in this prediction is to determine the vibration forcing functions to be used in the system and component dynamic analyses. In Subsection 3.9.2.3 of the SSAR, the applicant has outlined it's approach for determining these forcing functions. Because of the complexity of the flow conditions and structures involved, these loads are not determined by detailed analysis. Instead, a combination of analytical methods and predictions based on data from previously tested reactor internals of a similar design are used. This forcing function information is then used in a dynamic modal analysis to predict vibration amplitudes for each dominant response mode of components in the prototype ABWR and for interpretation of the preoperational and initial startup test results. Modal stresses are calculated and relationships are obtained between vibration measurement sensor responses and peak component stresses for each of the lower modes. The allowable amplitude in each mode is that which produces a peak stress amplitude of _ 10,000 psi. This stress is well below the allowable stress amplitude for cycles in excess of 10_ which is defined in the design fatigue curves for austenitic stainless steels in ASME, Section III, Appendix I.

As mentioned above, a reactor internals flow-induced vibration measurement and inspection program will be conducted on the first ABWR plant in conformance with the guidelines of Regulatory Guide 1.20 for prototype plants. These tests will be conducted in the following three phases:

(1) <u>Preoperational tests prior to fuel loading</u>. Steady-state test conditions will include balanced recirculation system operation and unbalanced operation over the full range of flow rates up to rated flow. Transient flow conditions will include single- and multiple pump trips from rated flow. This will subject major components to a minimum of 10_ cycles of vibration at the anticipated dominant response frequency and at the maximum response amplitudes. Vibration measurements will be obtained during this test and a close visual inspection of internals will be conducted before and after the test.

(2) <u>Precritical testing with fuel</u>. This vibration measurement series will be conducted with the reactor assembly complete but prior to reactor criticality. Flow conditions will include balanced, unbalanced, and transient conditions as for the first test series. The purpose of this series is to verify the anticipated effect of the fuel on the vibration response of internals.

(3) <u>Initial Startup testing</u>. Vibration measurements will be made during reactor startup at conditions up to 100% rated flow and power. Balance, unbalanced, and transient conditions of recirculation system operation will be evaluated. The primary purpose of this test series is to verify the anticipated effect of two-phase flow on the vibration response of internals.

Vibration sensor types may include strain gages, displacement sensors (linear variable transducers), and accelerometers. Accelerometers will be provided with double integration signal conditioning to give a displacement output. Sensor locations will include the following:

1. top of shroud head, lateral acceleration (displacement);

2. top of shroud, lateral displacement;

3. control rod drive housings, bending strain;

4. incore housings, bending strain; and

5. core flooder internal piping, bending strain.

In addition to these components, vibration of the core flooder sparger will be measured during preoperational testing of that system at the designated prototype plant.

In all prototype plant vibration measurements, only the dynamic component of strain or displacement is recorded. Data will be recorded on magnetic tape and provision made for selective online analysis to verify the overall quality and level of the data. Interpretation of the data requires identification of the dominant vibration modes of each component by the test engineer using frequency, phase, and amplitude information from the component dynamic analyses from which were discussed above. Comparison of measured vibration amplitudes to predicted and allowable amplitudes will then be made on the basis of the analytically obtained normal mode which best approximates the observed mode.

The visual inspections conducted prior to and following preoperational testing are for the purpose of detecting evidence of vibration, wear, or loose parts. At the completion of preoperational testing, the reactor vessel head and the shroud head will be removed, the vessel drained, and major components inspected. The inspections will cover the shroud, shroud head, core support structures, recirculation internal pumps, peripheral control rod drive, and incore guide tubes. Access is provided to the reactor lower plenum for these inspections.

A description of the vibration measurement and inspection phases of the above test program and a summary of the vibration analysis program should be submitted by the applicant for the staff's review in accordance with the schedules in Positions C.2.5.3 and C.2.5.4 of R.G. 1.20. A summary of the results of the vibration analysis, measurement and inspection programs should be submitted in accordance with the schedule in Position C.2.5.5 of R.G. 1.20.

In addition to a detailed discussion of the above information in Subsections 3.9.2.3, 3.9.2.4 and 3.9.2.4 and 3.9.2.6 of the SSAR, the applicant has further stated that ABWR plants which will be constructed subsequent to the ABWR prototype plant and which have reactor internals similar to those of the prototype plant will be tested in accordance with positions of R.G. 1.20 which are applicable to non-prototype plants.

On the basis of the above information, and its review of Subsections 3.9.2.3, 3.9.2.4 and 3.9.2.6 of the SSAR, the staff concludes that the applicant has met GDC 1 and 4 with respect to the reactor internals being designed and tested to quality standards commensurate with the importance of the safety functions being performed and being appropriately protected against dynamic effects (1) by meeting RG 1.20 for the conduct of preoperational vibration tests and (2) by having a preoperational vibration program planned for the reactor internals that provides an acceptable basis for verifying the design adequacy of these internals under test loading conditions comparable to those that will be experienced during operation. The combination of predictive analysis, pre-test inspections, tests and post-test inspection provides adequate assurance that the reactor internals will, during their service life, withstand the flow-induced vibrations of the reactor without loss of structural integrity. The integrity of the reactor internals in service is essential to ensure the proper positioning of reactor fuel assemblies and the incore instrumentation system to permit safe reactor operation and shutdown.

3.9.2.4 Dynamic System Analysis of Reactor Internals Under Faulted Conditions

Dynamic system analyses should be performed by the applicant to confirm the structural design adequacy, with no loss of function, of the reactor internals and unbroken loops of the reactor coolant piping to withstand the loads from a loss-of-coolant accident in combination with the safe shutdown earthquake. In Subsection 3.9.2.5 of the SSAR, the applicant has briefly described such an analysis. However, this Subsection and other referenced Subsections in the SSAR do not contain enough detailed information for the staff to evaluate. To comply with applicable portions of Standard Review Plan, Section 3.9.2, the staff must review the methods of analysis, the considerations used in defining the mathematical models, the descriptions of the forcing functions, the calculational scheme, the acceptance criteria and the interpretation of analytical results. In Question 210.33, the staff requested more detailed information. The applicant has not yet responded to this request. Subsequent to receipt of an acceptable response to Q 210.33, including a summary of results of the analyses to verify that the stresses and deformations are within allowable limits, the staffs evaluation in its final SER will be as follows:

On the basis of its review of Subsection 3.9.2.5 of the SSAR, the staff concludes that the applicant will meet applicable portions of GDC 2 and 4 and S.R.P. Section 3.9.2 by performing a dynamic system analysis that provides an acceptable basis for confirming the structural design adequacy of the reactor internals and unbroken piping loops to withstand the combined dynamic loads of a postulated LOCA (or applicable pipe rupture) and SSE. The analysis provides adequate assurance that the combined stresses and strains in the components of the reactor coolant system and reactor internals will not exceed the allowable design stress and strain limits for the materials of construction and that the resulting deflections or displacements at any structural element of the reactor internals will not distort the reactor internals geometry to the extent that core cooling may be impaired. The methods used for component analysis have been found to be compatible with those used for the system analysis. The combination of component and system analyses is, therefore, acceptable.

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

The staff's review under SRP Section 3.9.3 is concerned with the structural integrity and functional capability of pressure-retaining components, their supports, and core support structures that are designed in accordance with ASME Code Section III or earlier industrial standards. The staff has reviewed loading combinations and their respective stress limits, the design and installation of pressure-relief devices, and the design and structural integrity of ASME Code Class 1, 2, and 3 components and component supports.

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits

The staff has reviewed the methodology used for load combinations and the selected values of allowable stress limits. The applicant has evaluated all ASME Code Class 1, 2, and 3 components, component supports, core support components, control rod drive components, and other reactor internals using the load combinations and stress limits presented in Subsection 3.9.3.1 of the SSAR. The staff has reviewed this information and has concluded that it conforms to SRP Section 3.9.3 with the following exceptions:

1. In Question 210.34, the staff requested that the applicant provide justification for using ASME Level D Service Limits (Faulted Condition allowable stresses) in SSAR Table 3.9-2 for a loading combination which the staff considers to be Level B (Upset Condition Loads), i.e., normal loads plus the most limiting safety-relief valve loads plus turbine stop valve closure induced loads. The applicant has not yet responded to this Question.

2. In Question 210.42, the staff requested that the applicant provide the design basis which will be used to insure the structural integrity of safety-related heating, ventilation and air conditioning ductwork and its supports. The applicant has not yet responded to this Question.

3. In Question 210.50, the staff requested that the applicant 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. The applicant has not yet responded to this request.

4. The applicant is requested to identify any piping system, component or equipment in the ABWR Nuclear Island which is designed for a life expectancy greater than 40 years. If any of these components are classified as ASME Class 2 or 3 or Quality Group D, and are subjected to loadings which could result in thermal or dynamic fatigue, the applicant is further requested to performing fatigue analyses on these components which are similar to the requirements for Class 1 components in ASME III, Subsection NB. In addition to the transients discussed in Subsection 3.9.1 of the SSAR, the loadings for these analyses should account for operating vibration loads which may have been observed during piping preoperational tests and for the effects of mixing hot and cold fluids.

The ASME Code requires that a design specification be prepared for Class 1, 2, and 3 components such as pumps, valves, and piping systems. The design specification is intended to become a principal document governing design and construction of these components and should include specification of loading combinations, design data, and other design data inputs. The Code also requires a design report for ASME Code Class 1, 2, and 3 piping and components. During its review of the ABWR SSAR, the staff plans to audit and review design documents for selected pumps, valves, and piping systems to determine that the selected design specifications and design reports are in compliance with ASME Code requirements and are acceptable. The staff has not completed this review. Results of the design specification review will be submitted by the staff in the Final SER.

Subsequent to resolution of the issues discussed above, the staffs' evaluation of this section will be as follows:

On the basis of its review of Subsection 3.9.3.1 in the SSAR, the staff finds that the applicant has met 10 CFR 50.55a and GDC 1, 2, and 4 with respect to the design and service load combinations and associated stress and deformation limits specified for ASME Code Class 1, 2, and 3 components by ensuring (1) that systems and components important to safety are designed to quality standards commensurate with their importance to safety and (2) that these systems can accommodate the effects of normal operation as well as postulated events such as LOCAs and the dynamic effects resulting from earthquakes. The specified design and service combinations of loading as applied to ASME Code Class 1, 2, and 3 pressure-retaining components in systems designed to meet seismic Category I standards provide assurance that, in the event of an earthquake affecting the site or other service loading caused by postulated events or system operating transients, the resulting combined stresses imposed on system components will not exceed allowable stress and strain limits for the materials of construction. Limiting stresses under such loading combinations provides a conservative basis for the design of system components to withstand the most adverse combination of loading events without loss of structural integrity.

3.9.3.2 Design and Installation of Pressure-Relief Devices

The staff has reviewed Subsection 3.9.3.3 in the SSAR with respect to the design, installation, and testing criteria applicable to the mounting of pressure-relief devices used for the overpressure protection of ASME Code Class 1, 2, and 3 components. This review, conducted in accordance with SRP Section 3.9.3, includes evaluation of the applicable loading combinations and stress criteria. The design review extends to consideration of the means provided to accommodate the rapidly applied reaction force when a safety valve or relief valve opens, and the transient fluid-induced loads applied to the piping downstream of a safety or relief valve in a closed discharge piping system. The information in Subsection 3.9.3.3, Amendment 3 to the SSAR meets the applicable guidelines of SRP 3.9.3 and is therefore acceptable.

In accordance with Item II.D.1 of NUREG-0737 "Clarification of TMI Action Plan Requirements," pressurized water reactor and boiling water reactor licensees and applicants are required to conduct testing to qualify the reactor coolant system relief and safety valves, and associated piping and supports under expected operating conditions for design-basis transients and accidents. The applicant's response to Item II.D.1 is briefly discussed in Subsection 1A.2.9, Appendix 1A of the SSAR. This subsection states that the safety/relief valve models which will be used with ABWR plants are expected to be very similar to existing models which have undergone testing for alternate shutdown cooling mode flow conditions. The applicant is requested to submit a more detailed response for Item II.D.1 of NUREG-0737. This response should include a description of the test program which was conducted and the basis for concluding that the results of the test program envelopes all of the safety/relief valves and associated piping which will be used in the ABWR Nuclear Island.

Subsequent to receipt of acceptable information relative to Item II.D.1 of NUREG-0737, the staffs' evaluation in its final SER will be as follows:

On the basis of its review of Subsections 3.9.3.3 and 1A.2.9 of the SSAR, the staff finds that the applicant has met 10 CFR 50.55a and GDC 1, 2, and 3 with respect to the criteria to be used for design and installation of ASME Code Class 1, 2, and 3 overpressure-relief devices by ensuring that safety and relief valves and their installations will be designed to standards that are commensurate with their safety functions, and that they will accommodate the effects of discharge caused by normal operation as well as postulated events such as LOCAs and the dynamic effects resulting from the SSE. The applicant also has met GDC 14 and 15 with respect to ensuring that the reactor coolant pressure boundary design limits for normal operation, including anticipated operational occurrences, will not be exceeded. The criteria used by the applicant in the design and installation of ASME Code Class 1, 2, and 3 safety and relief valves provide adequate assurance that, under discharging conditions, the resulting stresses will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under the loading combinations associated with the actuation of these pressure-relief devices provides a conservative basis for the design and installation of the devices to withstand these loads without loss of structural integrity or impairment of the overpressure protection function.

3.9.3.3 Component Supports

The staff's review of Subsections 3.9.3.4 and 3.9.3.5 of the SSAR relates to the methodology used in the design of ASME Code Class 1, 2, and 3 component supports. The review includes assessment of design and structural integrity of the supports. The review addresses three types of supports: plate and shell, linear, and component standard types. All ASME Class 1, 2, and 3 component supports for the ABWR Nuclear Island will be constructed in accordance with the rules of ASME Section III, Subsection NF, "Component Supports." The staff finds this to be an acceptable commitment pending resolutions of the issues discussed below. Loading combinations for component supports are discussed in Section 3.9.3.1 of this SER.

The staff requests that the applicant's commitment to conform to the rules of ASME III, Subsection NF be augmented by providing the following information:

1. Provide the rules which will govern the design of single angle members of ASME Class 1, 2, 3 and MC linear component supports.

2. Provide the methodology to be used to account for warping stresses which may occur when open sections of Class 1, 2, 3 and MC linear component supports are subjected to torsionial loads.

In SSAR Question 210.39, the staff requested that the applicant 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. The applicant has not yet responded to this request.

In its response to Question 210.40, the applicant revised Subsections 3.9.3.4.2 and 3.9.3.5 in the SSAR to provide buckling criteria for the reactor pressure vessel support skirt and other ASME III component supports, respectively. The information in Subsection 3.9.3.4.2 is acceptable. In Subsection 3.9.3.5, the statement is made that supports are evaluated in accordance with ASME III. This statement is not completely acceptable. For these component supports, the staff will require more explicit buckling criteria similar to that in Subsection 3.9.3.4.2. In addition, Subsection 3.9.3.5 states that buckling limits not specifically enumerated by the ASME III will be based on the criteria in Subsections 3.9.3.5(1), (2) & (3). If the criteria in these three Subsections are applicable to plate and shell type component supports, the ASME Service Level D Limits do not agree with the staffs position and ASME Section III rules of using a maximum limit of two-thirds of the critical buckling load. The applicant is requested to clarify the response to Q 210.40 by addressing the above staff comments.

In the response to the staffs' Question 210.41, the applicant stated that for equipment mounted on a concrete support, sufficient holes for anchor bolts are provided to limit the bolt stress to less than 10,000 psi on the nominal bolt area in shear or tension. This criterion is not completely acceptable. The applicant is requested to revise its response to Q 210.41 to provide a commitment that concrete anchor bolts which are used for pipe support base plates will be designed to the applicable factors of safety which are defined in I&E Bulletin 79-02, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts," Revision 1 dated June 21, 1979.

Subsequent to resolution of the three issues discussed above, the staffs' evaluation in the final SER will be as follows:

On the basis of its review of Subsection 3.9.3.4 in the SSAR, the staff finds that the applicant has met 10 CFR 50.55a and GDC 1, 2, and 4 with respect to the design and service load combinations and associated stress and deformation limits specified for ASME Code Class 1, 2, and 3 component supports by ensuring (1) that component supports important to safety will be designed to quality standards commensurate with their importance to safety and (2) that these supports will accommodate the effects of normal operation as well as postulated events such as LOCAs and the dynamic effects resulting from the SSE. The combination of loadings (including system operating transients) considered for each component support within a system, including the designation of the appropriate service stress limit for each loading combination, will be in accordance with the Standard Review Plan, Section 3.9.3. The specified design and service loading combinations used for the design of ASME Code Class 1, 2, and 3 component supports in systems classified as seismic Category I provide assurance that, in the event of an earthquake or other service loadings caused by postulated events or system operating transients, the resulting combined stresses imposed on system components and component supports will not exceed allowable stress and strain limits for the materials for the materials of construction. Limiting the stresses under such loading combinations provides a conservative design basis to assure that support components will withstand the most adverse combination of loading events without loss of structural integrity.

Class CS component evaluation findings are addressed in SER Section 3.9.5.

3.9.4 Control Rod Drive Systems

The staff's review under SRP Section 3.9.4 includes the control rod drive system up to its interface with the control rods. Those components of the CRDS which are part of the primary pressure boundary are classified as Safety Class 1, Quality Group A and are designed according to ASME Section III Class 1 requirements. The control rod drive system (CRDS) shall be capable of reliably controlling reactivity changes either under conditions of anticipated normal plant operational occurrences or under postulated accident conditions. The CRDS in the ABWR consists of fine motion control rod drive mechanisms and the control rod drive hydraulic system. The staff reviewed the information in Subsections 3.9.4 and 4.6.3 of the SSAR relative to the analyses and tests performed to ensure the structural integrity and functional capability of this system during normal operation and under accident conditions. Loading combinations for the CRDS are discussed in Section 3.9.3.1 of this SSER.

In Question 210.43, the staff requested more information relative to the test programs which were conducted for the three different prototype designs of the fine motion control rod drive mechanism. The applicant has not yet responded to this request.

Subsequent to resolution of the two issues discussed above, the staff's evaluation in its final SER will be as follows:

On the basis of its review of Subsections 3.9.4 and 4.6.3 of the SSAR, the staff concluded that the design of the control rod drive system is acceptable for the ABWR and has met GDC 1, 2, 14, 26, 27, and 29 and 10 CFR 50.55a.

The applicant has met GDC 1 and 10 CFR 50.55a with respect to designing components important to safety to quality standards commensurate with the importance of the safety functions to be performed. The design procedures and criteria used for control rod drive systems are in conformance with appropriate ANSI and ASME codes.

The applicant has met GDC 2, 14, and 26 with respect to designing the control rod drive system to withstand the effects of earthquakes and anticipated normal operation occurrences with adequate margins to ensure its structural integrity and functional capability and with extremely low probability of leakage or gross rupture of the reactor coolant pressure boundary. The specified design transients, design and service loadings, combinations of loads, and limiting the stresses and deformations under such loading combinations are in conformance with the appropriate ANSI and ASME Codes and acceptable regulatory positions specified in SRP Section 3.9.3.

The applicant has met GDC 27 and 29 with respect to designing the control rod drive system to ensure its capability to control reactivity and cool the reactor core with appropriate margin, in conjunction with either the emergency core cooling system or the reactor protection system. The operability assurance program is acceptable with respect to meeting system design requirements in observed performance as to wear, functioning times, latching, and overcoming a stuck rod.

3.9.5 <u>Reactor Pressure Vessel Internals</u>

The staff's review under SRP Section 3.9.5 is concerned with the load combinations, allowable stress and deformation limits, and other criteria used in the design of the reactor internals. Subsection 3.9.5.3.5 in the SSAR states that the core support structures for the ABWR will be constructed in accordance with the rules of ASME, Section III, Subsection NG, "Core Support Structures." This commitment agrees with Standard Review Plan, Section 3.9.5 and is, therefore, acceptable.

Subsection 3.9.5.3.6 in the SSAR presents the design bases for safety class reactor internals other than the core support structures. The design criteria, loading conditions and analyses that provide the basis for the design of these structures meet the guidelines of ASME Subsection NG-3000. These components are constructed so as not to adversely affect the integrity of the core support structures as required by ASME Subsection NG-1122. The staff does not agree with the Safety and Quality Group classifications of some of these components. This issue is discussed in Section 3.2.2 of this Draft SER under the response to Questions 210.11 and 210.45.

Subsequent to the resolution of the quality group classification discussed above and in Section 3.2.2, the staff's conclusion in the final SER will be as follows:

On the basis of its review of Subsection 3.9.5 of the SSAR, the staff concludes that design of reactor internals for the ABWR is acceptable and meets GDC 1, 2, 4, and 10 and 10 CFR 50.55a.

The applicant meets GDC 1 and 10 CFR 50.55a with respect to designing the reactor internals to quality standards commensurate with the importance of the safety functions to be performed. The design procedures and criteria used for the reactor internals are in conformance with the requirements of Subsection NG of ASME Code Section III.

The applicant meets GDC 2, 4, and 10 with respect to designing components important to safety to withstand the effects of earthquake and the effects of normal operation, maintenance, testing, and postulated LOCAs with sufficient margin to ensure that their capability to perform their safety functions is maintained and the specified acceptance fuel design limits are not exceeded.

The specified design transients, design and service loadings, and combinations of loading as applied to the design of the reactor internals structures and components provide reasonable assurance that, in the event of an earthquake or of a system transient during normal plant operation, the resulting deflections and associated stresses imposed on these structures and components will not exceed allowable stresses and deformations under such loading combinations. This provides an acceptable basis for the design of these structures and components to withstand the most adverse loading events that have been postulated to occur during service lifetime without loss of structural integrity or impairment of function.

3.9.6 Inservice Testing of Pumps and Valves

The staff review under SRP Section 3.9.6 is concerned with the inservice testing of certain safety-related pumps and valves typically designated as ASME Code Class 1, 2, or 3. In Section 3.9.3 of this Draft SER, the staff discusses the design of safety-related pumps and valves in the ABWR Nuclear Island. The load combinations and stress limits used in the design of pumps and valves ensure that the component pressure boundary integrity will be maintained. In addition, the applicant will periodically test and perform measurements of all safety-related pumps and valves. These tests and measurements will be performed in accordance with Section XI of the ASME Code as required by 10 CFR 50.55a(g). The tests will verify that these pumps and valves operate successfully when called on. Periodic measurements of various parameters will be compared to baseline measurements to detect long-term degradation of the pump or valve performance. In Subsection 3.9.6 of the SSAR, the applicant has stated that details of the inservice test program, including test schedules and frequencies will be reported in the inservice inspection and testing plan. However, there is no mention of a schedule for submittal of this program. The staff will require that this program be submitted so that the review can be completed prior to Design Certification of the ABWR Nuclear Island. Subsection 3.9.6 should be revised to reflect such a commitment.

In Question 210.47, the staff requested a commitment that all safety-related pumps and valves should be included in the inservice testing program even if they are not categorized as ASME Class 1, 2, or 3. The applicant has not yet responded to this request.

Several safety systems connected to the reactor coolant pressure boundary have design pressure below the rated reactor coolant system (RCS) pressure. Also, some systems that are rated at full reactor pressure on the discharge side of pumps have pump suction below RCS pressure. To protect these systems from RCS pressure, two or more isolation valves are placed in series to form the interface between the high-pressure RCS and the low-pressure system. The leaktight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low-pressure systems. In Question 210.49, the staff requested a commitment from the applicant 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. The applicant has not yet responded to this request.

In Question 210.48 the staff requested that the applicant revise Subsection 3.9.6 to provide a more explicit commitment that the ABWR systems be designed to accommodate the applicable code requirements for inservice testing of pump and valves. In its response to this question the applicant removed the section of the SSAR dealing with relief requests and indicated that it was not their intention to take exception to Section XI requirements. This response is acceptable. However, as discussed above, the staff intends to perform a review of the inservice testing program to verify that system and component designs in the ABWR accommodate inservice testing.

Section XI contains requirements for the inservice testing of pumps and valves. The staff has determined that these requirements must be supplemented for the level of assurance of operability desired for the advanced light water reactor designs. The applicant is requested to provide a commitment to design and test the components as discussed below.

Pumps

Many currently operating plants perform pump testing on minimum flow recirculation loops. This type of testing provides little meaningful information on the operability of the pump and the cumulative effect of minimum flow testing may be damaging to the pumps. For ALWR designs, piping configurations are to be provided to accommodate inservice testing at a flow rate at least as large as the maximum design flow for the pump. In addition, the sizing of each minimum recirculation flow path must be evaluated to assure that its use under all analyzed conditions will not result in degradation of the pump. The flow rate through minimum recirculation flow paths is also to be periodically measured to verify that the flow is in accordance with the design specification.

For safety related pumps in ALWR designs, the pumps are to be provided with instrumentation to verify that the net positive suction head (NPSH) is greater than or equal to the NPSH required during all modes of pump operation.

All safety related pumps are to be periodically disassembled and inspected to determine if there are any indications of unacceptable wear. The frequency of inspection may vary depending upon the service of the pump. The applicant should include the proposed frequency for each pump in the commitment to disassemble and inspect the pumps.

Check Valves

Piping designs are to incorporate provisions for full flow testing to demonstrate the operability of the valves under design conditions.

Inservice testing is to incorporate the use of advanced non-intrusive techniques to periodically assess degradation and the performance characteristics of the valves.

In addition to the above testing, check valves are to be periodically disassembled and inspected to determine if there are any indications of unacceptable wear, corrosion, or other forms of degradation. The frequency of inspection may vary depending upon the service conditions. The applicant should include the proposed frequency for each valve or group of valves in the commitment on disassembly and inspection.

Motor Operated Valves (MOV)

The design is to address the concerns and issues identified in I&E Bulletin (IEB) 85-03, IEB 85-03 Supplement 1, and the forthcoming Generic Letter extending IEB 85-03. The commitment on this subject should specifically address the method of assessment of the loads, the method of sizing the actuators, and the setting of the torque and limit switches.

The design is to incorporate the results of either in-situ or prototype testing with full flow and pressure or full differential pressure to verify the proper sizing and correct switch settings of the valves. The design should include a study to determine the optimal frequency for valve stroking during inservice testing such that unnecessary testing and damage is not done to the valve as a result of the testing.

The inservice testing of MOVs is to rely on diagnostic techniques that are consistent with the state of the art, that are diagnostic of the condition of the valve, and that will permit an assessment of the performance of the valve under actual loading.

In addition to the above testing, MOVs are to be periodically disassembled and inspected to determine if there are any indications of unacceptable wear, corrosion, or other forms of degradation. The frequency of inspection may vary depending upon the service of the valve. The applicant is to include the proposed frequency for each valve or group of valves in the commitment on disassembly and inspection.

Check Valves and Motor Operated Valves

The applicant is to verify the leak tight integrity of each valve relied upon in the safety analysis to provide a leak tight function. The types of valves that are of particular interest to the staff are: pressure isolation valves- valves that provide isolation of pressure differential from one part of a system from another or between systems, temperature isolation valves - valves whose leakage may cause unacceptable thermal stress, fatigue or stratification in the piping and thermal loading on supports or whose leakage may cause steam binding of pumps, and containment isolation valves - valves that perform a containment isolation function including valves that may be exempted from Appendix J, Type C, testing but whose leakage may cause loss of water inventory of a suppression pool.

Subsequent to resolution of the issues discussed above, the staff's evaluation in the Final SER will be as follows:

On the basis of its review of Section 3.9.6 of the ABWR SSAR, the staff concludes that the applicant's commitment to a pump and valve inservice testing program is acceptable and meets the requirements of 10 CFR Part 50, Appendix A, General Design Criteria 37, 40, 43, 46, 54, and _ 50.55a(g). This conclusion is based on the applicant's commitments to provide a test program to ensure that safety-related pumps and valves will be in a state of operational readiness to perform necessary safety functions throughout the life of the plant. This program will include baseline preservice testing and periodic inservice testing of the components in the operational state. The applicant has also committed to include all safety-related Code Class 1, 2, and 3 pumps and valves and to include those pumps and valves which are not Code Class 1, 2, and 3 but are considered to be safety related.

3.10 Seismic and Dynamic Qualification of Safety-Related Mechanical and Electrical Equipment

In Subsection 3.9.2.2 and and Section 3.10 of the SSAR, the applicant has presented information relative to seismic and dynamic qualification of safety-related mechanical and electrical equipment. In Subsection 3.9.3.2, the applicant has also presented information relative to pump and valve operability assurance. This information included the following:

- Rationale used to determine whether tests, analyses or combinations of both will be performed.

- Criteria used to define the seismic and other relevant dynamic load input motions.

- The proposed demonstration of the adequacy of the qualification program.

The seismic qualification methodology presented in Section 4.4 of NEDE-24326-1-P, "General Electric Environmental Qualification Program" will be used by the applicant for both mechanical and electrical equipment. This program conforms to the requirements of IEEE-323, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations" as modified and endorsed by Regulatory Guide 1.89, "Qualification of Class 1E Equipment for Nuclear Power Plants." The program also meets the criteria contained in IEEE-344, "Guide for Seismic Qualification of Class 1E Electric Equipment for Nuclear Power Generating Stations," as modified by Regulatory Guide 1.100, "Seismic Qualification of Electric Equipment for Nuclear Power Plants." In Tables 1.8-20 and 1.8-21 of the SSAR, the applicant has agreed to use Regulatory Guide 1.100, Revision 2, June 1988 and IEEE-344, 1987. The staff reviewed NEDE-24326-1-P and approved the qualification methodology therein in an SER sent to General Electric on October 23, 1983. It is the staff's understanding that this methodology in NEDE-24326-1-P has been updated for the ABWR by the commitments in Tables 1.8-20 and 1.8-21 of the SSAR which are discussed above. If this is not the case, the staff requests the applicant to revise the SSAR to provide a commitment to update NEDE-24326-1-P. The staff considers this program to be applicable to mechanical as well as electrical equipment.

NEDE-24326-1-P presents qualification methodology only and contains no plant specific information. Therefore, each applicant referencing this document must insure that specific environmental parameters along with seismic and dynamic input response spectra are properly defined and enveloped in the methodology for its specific plant. In Subsection 3.9.3.2 and Section 3.10 of the SSAR, the applicant has committed to provide documentation of the results of both the pump and valve operability and the seismic and dynamic qualification programs. In accordance with applicable guidelines of Standard Review Plan, Section 3.10, the staff will conduct audits of the ABWR Nuclear Island files to review the results of tests and analyses which were performed to assure the proper implementation of criteria outlined in the SSAR, to assure that adequate qualification has been demonstrated for all equipment and their supports, and to verify that all applicable loads have been properly defined and accounted for in the testing/analyses performed. These audits have to be completed prior to Design Certification of the ABWR Nuclear Island. Therefore, the applicant will be requested to support such audits prior to the issuance of the staffs final SER. In addition, during construction of each ABWR plant, the staff will perform walkdowns of equipment components to verify conformance of their as-built configurations to those indicated in the qualification documentations.

In the response to Question 210.29, the applicant stated that in the seismic qualification of the control rod drive and CRD housing, the housing structural analysis is not included in the correlation of the CRD and fuel channel test results with analyses. The applicant is requested to provide the basis for this procedure.

Subsequent to successful completion of these audits, the staffs evaluation in the final SER will be as follows:

On the basis of the information in Subsections 3.9.2.2, 3.9.3.2 and Section 3.10 in the SSAR and the staffs audits of applicable ABWR files, the staff concludes that appropriate seismic and dynamic qualification and pump and valve operability program have been defined and substantially implemented. These programs meet applicable portions of GDC 1, 2, 4, 14 and 30 of Appendices A and B to 10 CFR 50 and Appendix A to 10 CFR 100.

ENCLOSURE 2

PRELIMINARY DRAFT SER, ABWR
STRUCTURAL AND GEOSCIENCES BRANCH

3.3 Wind and Tornado Loadings

3.3.1 Wind Design Criteria

All seismic Category I structures within the ABWR Standard Plant exposed to wind forces are designed to withstand the effects of the design wind. The design wind specified has a velocity of 130 mph based on a recurrence period of 100 years.

The procedures that are used to transform the wind velocity into pressure loadings on structures and the associated vertical distribution of wind pressures and gust factors are in accordance with ANSI A58.1 and ASCE paper 3269 (1961). These documents are acceptable to the staff.

The staff concludes that the plant design with respect to wind, along with the interface requirements mentioned at the end of this section, is acceptable and meets the requirements of GDC 2. This conclusion is based on the following.

GE has met the requirements of GDC 2 with respect to the capability of the structures to withstand design wind loading so that their design reflects

(1) appropriate consideration for the most severe wind not to exceed the velocity mentioned above for future site;

(2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena; and

(3) the importance of the safety function to be performed.

GE meets these requirements by using ANSI A58.1 and ASCE paper 3269, which the staff has reviewed and found acceptable, to transform the wind velocity into an effective pressure on structures and for selecting pressure coefficientscorresponding to the structures geometry and physical configuration.

GE designs the plant structures with sufficient margin to prevent structural damage during the most severe wind loadings that have been determined appropriate for the wind velocities mentioned above so that the requirements of item (1) listed above are met. In addition, the design of seismic Category I structures, as required by item (2) listed above, has included in an acceptable manner load combinations that occur as a result of the most severe wind load and the loads resulting from normal and accident conditions.

The procedures used to determine the loadings on structures induced by the design wind specified for the plant are acceptable because these procedures have been used in the design of conventional structures and proved to provide a conservative basis that, together with other engineering design considerations, ensures that, the structures will withstand such environmental forces.

The use of these procedures provides reasonable assurance that, in the event of design-basis winds, the structural integrity of the plant structures that have to be designed for the design wind will not be impaired and, in consequence, safety-related systems and components located within these structures are adequately protected and will perform their intended safety functions, if needed, thus satisfying the requirements of item (3) listed above.

Interface Requirements

The velocity of the wind as described above shall not be exceeded by the site-specific design-basis wind. This is to be ensured by an individual applicant. Also, any deviations submitted by an individual applicant from the plant arrangement of ABWR with respect to location or orientation of various buildings must be reviewed and accepted by the staff.

3.3.2 Tornado Design Criteria

All seismic Category I structures exposed to tornado forces and needed for the safe shutdown of the plant are designed to resist a tornado of 260-mph tangential wind velocity and 57-mph translational wind velocity. The simultaneous atmospheric pressure drop is assumed to be 1.46 psi at the rate of 0.27 psi per second. These tornado loadings are in accordance with ANSI/ANS 2.3. Tornado missiles are also considered in the design as discussed in Section 3.5 of this SER.

The considerations of tornado loadings in the design of seismic category I structures are in accordance with Bechtel Topical Report BC-TOP-3 which has been reviewed and approved by NRC staff for reference in plant applications. In BC-TOP-3, the procedures used to transform the tornado wind velocity into pressure loadings are similar to those used for the design wind loadings discussed in Section 3.3.1 of this report. The tornado missile effects are determined using procedures discussed in Section 3.5 of this SER. The total effect of the design tornado on seismic Category I structures is determined by appropriate combinations of the individual effects of the tornado wind pressure, pressure drop, and tornado-associated missiles.

The applicant meets the requirements of SRP Section 3.3.2 with respect to the structural capability to withstand design tornado wind loading and tornado missiles by:

(1) appropriate consideration of the most severe tornado not to exceed the tornado parameters mentioned above for future site.
(2) appropriate combinations of the effects of this severe natural phenomenon with those resulting from normal plant operation and/or accident conditions

(3) consideration of the importance of the safety function to be performed

The applicants will meet these requirements by using ANSI ANS 2.3 and BC-TOP-3 which the staff has reviewed and found acceptable.

GE designs the plant structures with sufficient margin to prevent structural damage during the most severe tornado loadings determined to be appropriate for most sites so that the requirements of Item (1) listed above will be met. In addition, the design of seismic Category I structures, as required by Item (2) listed above, includes, in an acceptable manner, load combinations involving the most severe tornado load and the loads resulting from normal plant operation and/or accident conditions. The procedures to determine the loadings on structures induced by the design-basis tornado specified for the plant have been used in the design of conventional structures and proven to provide a conservative basis, which, together with other engineering design considerations, will ensure that the structures will withstand such severe environmental forces.

The use of these procedures provides reasonable assurance that, in the event of a design-basis tornado, the structural integrity of the plant structures that have to be designed for the tornadoes will not be impaired and, consequently, safety-related systems and components located within these structures will be adequately protected and will perform their intended safety functions if needed, thus satisfying the requirement of Item (3) listed above.

Interface Requirements

The individual applicant shall ensure that tornado parameters as described above shall not be exceeded by those of the site-specific design basis tornado and that the collapse of nonseismic Category I structures, such as cooling towers or stacks outside the nuclear island, would not endanger seismic Category I struc- tures. Also, the future applicant must make a commitment that site-dependent effects of blast environmental loads are less than those of design tornado pressures or justify otherwise. The same commitment and/or justification is required for aircraft impact effects, as applicable.

3.4.2 Water Level (Flood) Design Procedure

For the ABWR Standard Plant structures, the design basis flood elevation is one foot (1 ft.) below grade and the design basis ground water is two feet below grade. Since the structures will be located above the flood elevation, no dynamic force on structures is considered. The lateral hydrostatic pressure and the hydrodynamic forces (due to earthquakes) on the structures due to the design flood water level as well as ground water are taken into consideration with other loads in load combinations specific in ABWR standard safety analysis report for structures.

On the basis of the review of the information provided the staff concludes that the plant design is acceptable and meets the recommendations of SRP Section 3.4.2 (NUREG-0800) and the requirements of GDC 2. This conclusion is based on the following:

GE has met the recommendations of SRP Section 3.4.2 and the requirements of GDC 2, with respect to the structural capability to withstand the effects of the flood or highest groundwater level, so that their design reflects

(1) appropriate consideration for the most severe flood not to exceed the flood level mentioned above for future site;

(2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena; and

(3) the importance of the safety function to be performed.

By limiting the design basis flood elevation to one foot below grade and the design basis ground water to two feet below grade in the design of plant structures, GE provides sufficient margin to prevent structural damage during the most severe flood or groundwater described above and the associated dynamic effects that have been determined appropriate for the flood levels so that the requirements of item (1) listed above are met. In addition, the design of seismic Category I structures, as required by item (2) listed above, includes in an acceptable manner load combinations that will occur as a result of the most severe flood or groundwater-related loads described above and the loads resulting from normal and accident conditions.

The procedures used to determine the loadings on seismic Category I structures induced by the design flood or highest groundwater level specified for the plant are acceptable because these procedures have been used in the design of conventional structures and proved to provide a conservative basis that, together with other engineering design considerations, ensures that the structures will withstand such environmental forces.

The use of these procedures provides reasonable assurance that, in the event of floods or high ground water as described above, the structural integrity of the plant seismic Category I structures will not be impaired and, in consequence, seismic Category I systems and components located within these structures will be adequately protected and may be expected to perform necessary safety functions, as required, thus satisfying the requirement of item (3) listed above.

Interface Requirements

The specific description of the individual applicant's site and elevations for all safety-related, structures, exterior accesses, equipment, and systems, from the standpoint of hydrology considerations and flood history (including date, level, peak discharge and related information for major historical flood events in the site region) will be provided by the individual applicant. The following topics will also be addressed by the individual applicant:

(1) probable maximum precipitation (PMP)

(2) precipitation losses

(3) runoff and stream course models

(4) maximum flood flow

(5) water level determination

(6) coincident wind wave activity

The individual applicant shall ensure and demonstrate in his plant-site-unique application that all the seismic Category I structures are either protected against flood damage or are not subject to damage by flooding. Hydrostatic and hydrodynamic effects of the flood are considered and described for all postulated design flood levels for the conditions set for the future site as outlined above.

3.5.3 Barrier Design Procedures

The plant's seismic Category I structures, systems, and components are to be shielded from, or designed for, various postulated missiles. Missiles considered in the design of structures include tornado-generated missiles.

The procedures used in the design of the structures, shields, and barriers to resist the effect of the missiles have been reviewed and found to be adequate. The analyses of structures, shields, and barriers to determine the effects of missile impact are accomplished in two steps. In the first step, the potential damage that could be done by the missile in the immediate vicinity of impact is investigated. This is accomplished by estimating the depth of penetration of the missile into the impacted structure. Furthermore, secondary missiles are prevented by fixing the target thickness well above that determined for penetration. In the second step of the analysis, the overall structural response of the target when impacted by a missile is determined using established methods of impactive analysis. The equivalent loads of missile impact, whether the missile is environmentally generated or accidentally generated within the plant, are combined with other applicable loads as discussed in Section 3.8 of this SER.

The staff concludes that the barrier design is acceptable and meets the recommendations of SRP Section 3.5.3 GDC 2 and 4 with respect to the capabilities of the structures, shields, and a barriers to provide sufficient protection to equipment that must withstand the effects of natural phenomena (tornado missiles). This conclusion is based on the following:

The procedures used for determining the effects and loadings on seismic Category I structures and missile shields and barriers induced by design-basis missiles selected for the plant are acceptable, since these procedures provide a conservative basis for engineering design to ensure that the structures or barriers will be adequately resistant to withstand the effects of such forces.

The use of these procedures will provide reasonable assurance that if design-basis missiles should strike seismic Category I structures or other missiles shields and barriers, the structures, sheilds, and barriers will not be impaired or degraded to an extent that will result in a loss of required protection. Seismic Category I systems and components protected by these structures will, therefore, be adequately protected against the effects of missiles and will perform their intended safety function, if needed. Conformance with these procedures is an acceptable basis for satisfying in part the requirements of GDC 2 and 4.

Interface Requirement

The individual applicant shall ensure and demonstrate that the tornado missiles and their associated impacting effects based on site-dependent parameters are less than those considered in ABWR, or justify the exceedances, for the staff's review and approval.

3.7 Seismic Design

3.7.1 Seismic Input

The input seismic design response spectra for operating basis earthquake (OBE) and safe shutdown earthquake (SSE) are defined at the ground surface. These spectra comply with RG 1.60. The maximum horizontal as well as the maximum vertical ground acceleration is 0.3g for the safe-shutdown earthquake (SSE). The corresponding peak acceleration for the operating-basis earthquake (OBE) is 0.15g which was used for the design of structures and components. The synthetic time history used for seismic design of ABWR seismic Category I structures, systems, and components is adjusted in amplitude and frequency to obtain response spectra that envelop the ABWR OBE design response spectra defined by RG 1.60, normalized to maximum ground accelerations. The magnitude of the SSE design time history is equal to twice the magnitude of the design OBE time history.

In the design of structures the above design time history is applied in the free-field at the grade level.

The damping ratios (expressed as a percentage of critical) used in the analysis of ABWR seismic Category I structures, systems, and components are in compliance with those listed in RG 1.61. For soils, damping values are determined on the basis of the soil shear strains induced in the free field.

The following ABWR seismic Category I structures have concrete mat foundations supported on soil, rock, or compacted backfill. The embedment depth from the plant grade to the bottom of the base mat is given for each seismic Category I structure as follows:

(1) reactor building (shield building, containment vessel, drywell, and reactor pedestal)--85 ft.

(2) control building--40 ft.

(3) radwaste building-substructure--21 ft 4 in.

All of these buildings are designed to have independent foundations. In all cases, the above value of embedment is used for the dynamic analysis to determine the seismic soil-structure interaction (SSI) effects.

The following conclusions are subject to the outcome of the staff's audit of the seismic analysis results.

GE meets (subject to the following interface requirement) the relevant requirements of GDC 2 and Appendix A to 10 CFR 100 by appropriate consideration of the most severe earthquake to which the ABWR Category I structures, systems, and components will be subjected, including appropriate margin and considerations for two levels of earthquake [i.e., the safe shutdown earthquake (SSE) and operating basis earthquake (OBE)]. GE has met these requirements by the use of the following methods and procedures.

The seismic design response spectra (OBE and SSE) used in the design of ABWR seismic Category I structures, systems, and components comply with the recommendations of RG 1.60. The specific percentage of critical damping values used in the seismic analysis of ABWR Category I structures, systems, and components are in conformance with RG 1.61, "Damping Values for Seismic Analysis of Nuclear Power Plants." The artificial synthetic time history used for seismic design of ABWR Category I structures, systems, and components is adjusted in amplitude and frequency content to obtain response spectra that envelop the design response spectra specified for the site. Conformance with the recommendations of RGs 1.60 and 1.61 ensures that the seismic inputs to the ABWR Category I structures, systems, and components are adequately defined to form a conservative basis for the design of such structures, systems, and components to withstand seismic loadings.

Interface Requirements

For specific future application it must be demonstrated that the site-specific design response spectra are less than or equal to those given in ABWR SSAR and in Section 2.5.2 of this SER which are in conformance with RG 1.60 normalized to the SSE and OBE peak ground accelerations.

3.7.2 Seismic System and Subsystem Analysis

The scope of review of the seismic system and subsystem analysis for the plant has included the seismic analysis methods for ABWR seismic Category I structures, systems, and components. It has included review of procedures for modeling, seismic soil-structure interaction, development of floor response spectra, inclusion of torsional effects, evaluation of seismic Category I structure overturning, and determination of composite damping. The review has included design criteria and procedures for evaluation of effects of parameter variations on floor response spectra. The review has also included criteria and seismic analysis procedures for seismic Category I buried piping and tunnels.

The system and subsystem analyses have been performed by GE on an elastic basis. Modal response spectrum and time history methods form the basis for the analyses of all major ABWR seismic Category I structures, systems, and components. When the modal response spectrum method is used, governing response parameters were combined by the method delineated in RG 1.92. Floor spectra inputs used for design and test verifications of structures, systems, and components are generated from the time history methods, taking into account variation of parameters by peak widening. A vertical seismic system dynamic analysis is used for all ABWR structures, systems, and components. Torsional effects and stability against overturning are considered.

A total of 14 site conditions with soil profile depths ranging from 85 feet to 300 feet and soil layer average shear wave velocities varying from 994 ft/sec to 10,000 ft/sec which represent sites of soft, median and stiff-soil conditions and rock sites are used in 42 cases of soil-structure interaction studies using the finite element computer program SASSI. These cases are identified in ABWR SSAR Table 3.A.7-1. Six (representative of three soil sites) of the 42 cases are further analyzed for soil-structure and structure-structure interaction using the CLASSI/ASD computer program which is based on continuum independence approach considering the soil medium as semiinfinite half space. The enveloped results of the six cases were evaluated against the all-site enveloped loads obtained from the 42 cases from the finite-element approach to establish seismic design loads for the structures. In both the finite element method (SASSI) and the half-space approach (CLASSI/ASD), GE used a deconvolution analysis to obtain the equivalent linear properties for shear modulus and material damping compatible with seismic strains induced in the free field.

Pending the satisfactory audit of the results of the analyses, the staff considers GE has fulfilled the SRP seismic design acceptance requirement by designing seismic Category I structures to responses validated by using two approaches of SSI analysis.

The staff concludes that the plant design is acceptable and meets the requirements of GDC 2 and Appendix A to 10 CFR 100. This conclusion is based on the following and subject to the confirmation of the items discussed above.

GE meets the requirements of GDC 2 and Appendix A to 10 CFR 100 with respect to the capability of the structures to withstand the effects of the earthquakes so that their design reflects:

(1) appropriate consideration for the most severe earthquake recorded for the site with an appropriate margin (GDC 2), consideration of two levels of earthquakes (Appendix A, 10 CFR 100);

(2) appropriate combination of the effects of normal and accident conditions with the effects of the natural phenomena; and

(3) the importance of the safety functions to be performed (GDC 2) (the use of a suitable dynamic analysis or a suitable qualification test to demonstrate the structures, systems, and components can withstand the seismic and other concurrent loads, except where it can be demonstrated that the use of an equivalent static load method provides adequate consideration (Appendix A, 10 CFR 100)).

GE has met the requirements of item (1) by use of the acceptable seismic design parameters in accordance with SRP Section 3.7.1. The combination of earthquake resulting loads with those resulting from normal and accident conditions in the design of seismic Category I structures as specified in SRP Section 3.8.1 through 3.8.5 will be in conformance with item (2).

The staff concludes that the use of the seismic structural analysis procedures and criteria delineated above by GE provides an acceptable basis for the seismic design that are in conformance with the requirements of item (3) listed above.

Interface Requirements

The following conditions are to be satisfied by individual applicants referencing ABWR:

(1) The peak ground acceleration is less than or equal to 0.3g SSE, 0.15g OBE as indicated in Section 2.5.2 of this SER.

(2) The site design ground response spectra are less than or equal to those given in RG 1.60 normalized to the peak ground accelerations in (1).
(3) There is no potential for liquefacation at the plant site resulting from OBE and SSE.

(4) There is no potential for fault displacement at the plant site.

(5) The embedment depths of the seismic Category I structures should be those mentioned in Section 3.7.1 of this SER.

(6) The average shear wave velocity of soil is 994 fps minimum. The upper bound shear wave velocity is 10,000 ft/sec.

(7) For layered soil sites with parameters that have very abrupt variations with depth, analysis with site-unique properties should be performed to confirm the applicability of the generic analysis.

(8) The soil-bearing capacity at the site is adequate to accommodate plant design loads.

In addition to the above conditions, at any site where the ABWR design is to be used, the staff will require that site-specific geotechnical data be developed by the individual applicants and submitted for review by the staff to demonstrate comparability with the design analyses assumptions (see Section 2.5.1 of this SER). The eight site-dependent conditions described above have to be satisfied by the individual applicant. In addition, the geotechnical parameters of future sites should be developed and reviewed by the staff with respect to those used in the ABWR seismic analyses to establish comparability. The fundamental frequencies of ABWR structures, equipment, and components will be limited to remain above a low frequency range (i.e., 4 Hz or other modified frequency range that may result from the eight additional analyses). Any deviation from this limitation will have to be justified and reviewed by the staff.

3.7.3 Seismic Instrumentation Program

The type, number, location, and use of strong-motion accelergraphs to record seismic events and to provide data on the frequency, amplitude, and phase relationship of the seismic response of the containment structure comply with RG 1.12 and SRP Section 3.7.4. Supporting instrumentation will be installed on ABWR seismic Category I structures, systems, and components to provide data for the verification of the seismic responses determined analytically for such seismic Category I items.

Subject to clarification of some of the applicant's statements the staff concludes that the seismic instrumentation system to be provided for the plant is acceptable and meets the requirements of GDC 2, 10 CFR 100, Appendix A and 10 CFR 50.55a. This conclusion is based on the following.

GE has met the requirements of 10 CFR 100, Appendix A, by providing the instrumentation that is capable of measuring the effects of an earthquake which meets the requirements of GDC 2. GE has met the requirements of 10 CFR 50.55a by providing the inservice inspection program that will verify operability by performing channel checks, calibrations, and functional test at acceptable intervals. In addition, the installation of the specified seismic instrumentation in the reactor containment structure and other ABWR seismic Category I structures, systems, and components constitutes an acceptable program to record data on seismic ground motion as well as data on the frequency and amplitude relationship of the seismic response of major structures and systems. A prompt readout of pertinent data at the control room can be expected to yield sufficient information to guide the operator on a timely basis for the purpose of evaluating the seismic response of major structures and systems in the event of an earthquake. Data obtained from such installed seismic instrumentation will be sufficient to determine that the seismic analysis assumptions and the analytical model used for the design of ABWR are adequate and that allowable stresses are not exceeded under conditions in which continuity of operation is intended. Provision of such seismic instrumentation complies with RG 1.12.

Interface Requirement

With the continuous enhancement in the state-of-art of seismic instrumentation, conformity with instrumentation guidelines existent at the time of individual license application will be required.

3.8.1 and 3.8.2 <u>Concrete Containment and it's Steel Components</u>

The containment is a reinforced concrete cylindrical shell structure with an internal steel liner which is mainly of carbon steel except for wetted surfaces where stainless steel or carbon steel with stainless steel cladding is used. It is divided by the diaphragm floor and the reactor pedestal into an upper and a lower drywell chamber and a suppression chamber. The containment is surrounded by and structurally integral with the reactor building. The containment wall is 6 feet 7 inches thick with an inside radius of 47 feet 7 inches and height of 96 feet 9 inches. The containment design pressure is 45 psig. The contain- ment is designed to resist various combinations of dead loads, live loads, environmental loads including those due to wind, tornadoes, and earthquakes, normal operating loads and loads generated by postulated LOCA. The concrete containment will be designed, fabricated, constructed and tested in accordance with subsection CC of the ASME Code Section III Division 2.

In order to verify the ultimate capacity of the concrete portion of the containment structure, a 1/6-scale global model was tested and the test results demonstrated that the ABWR concrete containment of a prototypical design is capable of withstanding 90 psig (2 times design pressure) without loss of structural integrity. The model was tested without subjection to high temperature. However, the pressure capacity of the concrete containment in combination with high temperature is expected to be much higher than 90 psig.

The major steel components of the concrete containment consist of: personnel airlocks, equipment hatches, and drywell head. These components will be designed for the same loads and load combinations as used in the design of the concrete containment shell to which these components are attached. These components are designed, fabricated and tested as class MC components in accordance with subsection NE of the ASME Code Section III Division 1.

Pending the provision of details of the containment steel liner design, the staff concludes that the design of the concrete containment and its associated steel components will be acceptable and will meet the requirements of SRP Sections 3.8.1 and 3.8.2 and relevant requirements of 10 CFR 50.34, GDC 1, 2, 4, 16 and 50. The conclusion is based on the following:

(1) The applicant will meet the recommendations of SRP Sections 3.8.1 and 3.8.2 and the requirements of GDC 1 by ensuring that the concrete containment and its steel components will be designed, fabricated, constructed, tested and inspected to the quality standards as stipulated in codes and regulatory guides.

(2) The applicant will meet the requirements of GDC 2 by designing the containment to withstand an earthquake that will envelop the most severe earthquake considered for prospective sites, and the combination of the effects of normal and accident conditions with the effects of environmental loadings such as earthquakes and other natural phenomena.

(3) The applicant will meet the requirements of GDC 4 by ensuring that the design of the containment will be such that it will be capable of withstanding the dynamic and thermal effects associated with missiles, pipe whipping, safety/relief valve discharge and discharging fluids resulting from events outside the nuclear power unit or from equipment failures.

(4) The applicant will meet the requirements of GDC 16 by having the containment so designed that it essentially will be a leaktight barrier to prevent the uncontrolled release of radioactive effluents to the environment.

(5) The applicant will meet the requirements of GDC 50 by designing the containment to accommodate, with sufficient margin, the design leakage rate and the calculated pressure and temperature conditions resulting from accident conditions, and by ensuring that the design conditions will not be exceeded during the full course of the accident condition. In meeting these design requirements, the applicant will use the recommendations of regulatory guides and industry standards. The applicant also will perform appropriate analysis that will demonstrate that the ultimate capacity of the containment will not be exceeded and will establish a reasonable margin of safety for the design.

The criteria used in the analysis, design, and construction of the containment structure to account for anticipated loadings and postulated conditions that may be imposed on the structure during its service lifetime are in conformance with established criteria, codes, standards, and guides and the industry standard, ASME Code, Section III, Division 1, Subsection NE, and Division 2.

The use of these criteria as defined by applicable codes, standards, and guides; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control programs; and the testing and inservice surveillance requirements will provide reasonable assurance that, in the event of earthquakes and various postulated accidents occurring within and outside the containment, the structure will withstand the specified conditions without impairment of structural integrity or safety function.

3.8.3 Concrete and Structural Steel Internal Structures

The main internal structures inside the containment are the reinforced concrete diaphragm, the reactor pedestal and reactor shield wall. The diaphragm separates the upper drywell from the suppression pool. The reactor pedestal consists of a ledge on a cylindrical shell which forms the reactor cavity extending from the bottom of the diaphragm to the top of the containment foundation slab. The space enclosed by the cylindrical shell under the reactor is the lower drywell which is connected to the suppression pool through a series of vertical and horizontal vents in the shell wall. A steel equipment platform is located in the lower drywell and is accessible through a steel personnel tunnel and a steel equipment tunnel from outside of the containment. Other internal structures are drywell equipment and pipe support structure, miscellaneous floors, and reactor shield wall stabilizer. The major code used in the design of concrete internal structures, is ACI 349 code. For all steel internal structures, the ANSI/AISCN 690 "Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities," is used. (For equipment supports, Subsection NF of the ASME code is used.)

The containment concrete and steel internal structures are designed to resist various combinations of dead and live loads, accident-induced loads (including pressure and jet loads), and seismic loads. The load combinations used cover those cases likely to occur and include all loads that may act simultaneously. The containment internal structures are designed and proportioned to remain within limits established by the staff under the various load combinations. These limits are, in general, based on the ASME Code Section III Division 2, on the ACI 349 code, and on the ANSI/AISCN 690 Specification for concrete and steel structures, respectively, modified as appropriate for load combinations that are considered extreme.

The design and analysis procedures that are used for the internal structures are the same as those that have been used for previously licensed applications and, in general, are in accordance with procedures delineated in the codes mentioned above.

The materials of construction, their fabrication, construction, and installation, are in accordance with the ACI 349 code and the ANSI/AISCN 690 specification for concrete and steel structures, respectively with the exception of the concrete diaphragm floor, for which ASME Code, Section III, Division 2 is used.

On the basis of review of the information provided, the staff concludes that the design of the containment internal structures is acceptable and meets the recommendations of SRP Section 3.8.3 and the relevant requirements of 10 CFR 50.55a, and GDC 1, 2, 4, and 50. This conclusion is based on the following:

(1) GE has met the recommendations of SRP Section 3.8.3 and the requirements of 10 CFR 50.55a and GDC 1 with respect to ensuring that the containment internal structures are designed, fabricated, erected, contracted, tested, and inspected to quality standards commensurate with the safety function to be performed by meeting the guidelines of regulatory guides and industry standards indicated below.

(2) GE has met the requirements of GDC 2 by designing the containment internal structure to withstand the most severe earthquake that has been established for the design envelope of ABWR with sufficient margin and the combination of the effects of normal and accident conditions with the effects of environmental loadings, such as earthquakes and other natural phenomena.

(3) GE has met the requirements of GDC 4 by ensuring that the design of the internal structures is capable of withstanding the dynamic effects associated with missiles, pipe whipping, and discharging fluids from safety/relief valves or equipment failure.

(4) GE has met the requirements of GDC 50 by designing the containment internal structures to accommodate, with sufficient margin, the calculated pressure and temperature conditions resulting from accident conditions, and by ensuring that the design conditions are not exceeded during the full course of the accident condition. In meeting these design requirements, the appli- cant has used the recommendations of regulatory guides and industry standards indicated below.

The criteria used in the analysis, design, and construction of the containment internal structures to account for anticipated loadings and postulated conditions that may be imposed on the structures during their service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the staff. These include meeting the positions of RGs 1.94, 1.136 and 1.142, and industry standards ACI 349, ASME Code, Section III, Division 2, "Code for Concrete Reactor Vessels and Containments," and AISC and ANSI/AISCN 690 "Specifications for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities."

The use of these criteria as defined by applicable codes, standards, guides, and specifications (on the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control programs, and the testing requirements) provide reasonable assurance that in the event of earthquakes and various postulated accidents occurring within the containment, the interior structures will withstand the specified design condi- tions without impairment of structural integrity or the performance of required safety functions.

3.8.4 Other Seismic Category I Structures

Seismic Category I structures within the ABWR design scope in the reactor building and the control building. (The radwaste building substructure is to remain intact under a seismic event to help contain liquid from a possibly ruptured tank. Overall, the radwaste building has no safety-related equipment; thus, the remainder of the building is not seismic Category I.)

Seismic Category I structures within ABWR design scope are constructed of structural steel and concrete. The structural components consist of slabs, walls, beams, and columns. The major code used in the design of concrete seismic Category I structures is the ACI 349, "Code Requirements for Nuclear Safety Related Concrete Structures." For steel seismic Category I structures, the ANSI/AISCN 690 "Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities" is used.

The concrete and steel seismic Category I structures within ABWR are designed to resist various combinations of dead loads; live loads; environmental loads including winds, tornadoes, OBE and SSE; and loads generated by postulated ruptures of high-energy pipes (such as reaction and jet impingement forces, compartment pressures, and impact effects of whipping pipes).

The design and analysis procedures that are used for these seismic Category I structures are the same as those approved on previously licensed applications and, in general, are in accordance with procedures delineated in the ACI 349 code and in the ANSI/AISCN 690 specification for concrete and steel structures, respectively.

The seismic Category I structures within ABWR Standard Plant are designed and proportioned to remain within limits established by the staff under the various load combinations. These limits are, in general, based on the ACI 349 code and on the ANSI/AISCN 690 specification for concrete and steel structures, respectively, modified as appropriate for load combinations that are considered extreme.

The materials of construction, their fabrication, construction, and installation, are in accordance with the ACI 349 code and the ANSI/AISCN 690 specification for concrete and steel structures, respectively.

Since the reactor building (RB) enclosures the containment and is integral with it, the effect of the hydrodynamic load on RB due to SRV discharge or LOCA in the containment should be taken into consideration. Pending the satisfactory review of this effect on RB the staff concludes that the design of safety-related structures within ABWR other than containment or containment interior structures is acceptable and meets the recommendations of SRP Section 3.8.4, the relevant requirements of 10 CFR 50.55a, and GDC 1, 2, 4, and 5. This conclusion is based on the following:

(1) GE has met the recommendations of SRP Section 3.8.4 and the requirements of 10 CFR 50.55a and GDC 1 with respect to ensuring that the safety-related structures other than containment are designed, fabricated, erected, contracted, tested, and inspected to quality standards commensurate with its safety function to be performed by meeting the guidelines of regulatory guides and industry standards indicated below.

(2) GE has met the requirements of GDC 2 by designing the safety-related structures other than containment to withstand the most severe earthquake that has been established for ABWR design envelope with sufficient margin and the combination of the effects of normal and accident conditions with the effects of environmental loadings, such as earthquakes and other natural phenomena.

(3) GE has met the requirements of GDC 4 by ensuring that the design of the safety-related structures are capable of withstanding the dynamic effects associated with missiles, pipe whipping, and discharging fluids.

(4) GE has met the requirements of Appendix B because the quality assurance program provides adequate measures for implementing guidelines relating to structural design audits.

The criteria used in the analysis, design, and construction of the ABWR seismic Category I structures to account for anticipated loadings and postulated conditions that may be imposed on each structure during its service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the staff. These include meeting the positions of RGs 1.28, 1.76, and 1.142, and industry standards ACI 349 and ANSI/ANSCN 690 "Specifications, for the Design, Fabrication, and Erection of Structural Steel Safety-related structures for Nuclear Facilities".

The use of these criteria as defined by applicable codes, standards, guides, and specifications (on the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, and quality control programs) provide reasonable assurance that, in the event of winds, tornadoes, earthquakes, and various postulated accidents occurring within the structures, the structures will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions.

Interface Requirements

Other seismic Category I structures not within the ABWR scope will be identified and described by individual applicants and their design and analysis will have to be reviewed and accepted by the staff on a case-by-case basis. The individual applicant should ensure that the settlements of adjacent buildings should be such that the integrity of underground piping or tunnel will not be jeopardized.

3.8.5 Foundations

In the ABWR design separate reinforced concrete mat foundations are considered for major seismic Category I structures. The reactor building foundation which is integral with the containment foundation supports the containment structure, the reactor pedestal and other internal structures. Even though the containment structure foundation is integral with the reactor building foundation, it is defined by GE as the portion of the foundation within the perimeter of the containment structure. As such it is designed as a part of the containment boundary. Primarily, these foundations are reinforced concrete of the mat type. The major code used in the design of these concrete mat foundations is ACI 349 except for the portion of the foundation within the containment boundary for which ASME Section III Division 2 code is used. These concrete foundations have been designed to resist various combinations of dead loads, live loads, environmental loads (including winds, tornadoes, OBE, and SSE), and loads generated by postulated ruptures of high-energy pipes. Detailed design information such as the factor of safety against flotation (buoyancy) for reactor building is calculated and provided. However there is no such information given for control building and radwaste building.

The design and analysis procedures that were used for these seismic Category I foundations are, in general, in accordance with procedures delineated in the ACI 349 and ASME Section III Division 2 codes. The various seismic Category I foundations were designed and proportioned to remain within limits established by the regulatory staff under the various load combinations. These limits are, in general, based on the ACI 349 and the ASME Section III Division 2 codes modified as appropriate for load combinations that are considered extreme. The materials of construction, their fabrication, construction and installation, will be in accordance with the ACI 349 and ASME Section III Division 2 codes. The criteria that were used in the analysis, design, and construction of all the ABWR seismic Category I foundations to account for anticipated loadings and postulated conditions that may be imposed on each foundation during its service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the NRC staff.

Pending the review of the detailed design information for the control building and radwaste building the staff concludes that the design of the seismic Category I foundations within ABWR are acceptable and meets recommendations of SRP Section 3.8.5 and the relevant requirements of 10 CFR 50.55a, and GDC 1, 2, 4, and 5. This conclusion is based on the following:

(1) GE meets the requirements of 10 CFR 50.55a and GDC 1 with respect to ensuring that the seismic Category I foundations within ABWR are designed, and will be fabricated, erected, constructed, tested, and inspected to quality standards commensurate with its safety function to be performed by meeting the guidelines of regulatory guides and industry standards indicated below.

(2) GE meets the requirements of GDC 2 by designing the seismic Category I foundation within the ABWR to withstand the most severe earthquake that has been established for ABWR design envelope with sufficient margin and the combination of the effects of normal and accident conditions with the effects of environmental loadings such as earthquakes and other natural phenomena.

(3) GE meets the requirements of GDC 4 by ensuring that the design of seismic Category I foundations within ABWR will be capable of withstanding the dynamic effects associated with missiles, pipe whipping, and discharging fluids.

(4) GE meets the requirements of GDC 5 by demonstrating that structures, systems, and components are not shared between units or that, if shared, sharing will not impair their ability to perform their intended safety function.

The criteria used in the analysis, design, and construction of all the ABWR seismic Category I foundations (to account for anticipated loadings and postulated conditions that may be imposed on each foundation during its service lifetime) are in conformance with established criteria, codes, standards, and specifications acceptable to the staff. These include meeting the positions of RG 1.142, industry standard ACI 349 ASME Section III Division 2 codes, and AISC "Specification for Design, Fabrication, and Erection of Structural Steel for Building."

The use of these criteria as defined by applicable codes, standards, guides, and specifications (on the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, and quality control programs, provide reasonable assurance that, in the event of winds, tornadoes, earthquakes, and various postulated events, seismic Category I foundations within ABWR withstand the specified design conditions without impairment of structural integrity and stability or the performance of required safety functions.

Interface Requirements

1. The individual applicant must ensure that the site-specific soil parameters and the settlement of foundations and structure evaluated therewith are comparable with the soil parameters used in the ABWR foundation design. Otherwise, the individual applicant must identify and justify the noncomparable aspects of the parameters. The staff will review the applicant's justification on a case-by-case basis.

2. If foundation waterproofing is used, the individual applicant should evaluat the capability of the foundations to transfer shear loads. The staff will review the applicant's evaluation.