20.2 Questions

This subsection provides an up-to-date chapter-wise listing of the NRC questions. Subsections are numbered (e.g., 20.2.x) in accordance with the questions received for specific chapters.

20.2.1 Chapter 1 Questions

420.116

The forth paragraph seems to imply that all three systems are needed to mitigate a LOCA. Is that accurate? (1.2.2.4.8.1.2)

430.227

Regarding TMI Action Item III.D.1.1 (NUREG-0737) concerning the integrity of systems outside containment likely to contain radioactive material for pressurized water reactors and boiling water reactors, provide information on the following items: (1A.2.34)

430.227a

Clarify whether the systems that require periodic leak testing listed in ABWR SSAR Subsection 1A.2.34 include systems unique to the ABWR design. Include such systems if they are not currently included in Subsection 1A.2.34. Also, include containment and reactor coolant sampling systems to the above list.

430.227b

Since ABWR SSAR Section 5.2.5 discusses leak detection methods outside primary containment which include secondary containment, turbine building and steam tunnel, rewrite Subsection 1A.2.34 to include all the areas mentioned above (current write-up refers to secondary containment only).

430.227c

SSAR Subsection 1A.2.34 states that all lines which pass outside the secondary containment contain leakage control systems or loop seals and that these systems are discussed in SSAR Section 6.5.3. However, these systems, particularly, the loop seal systems for the secondary containment penetrations, are not discussed in the SSAR Section 6.5.3. Discuss the above systems.

430.227d

SSAR Subsection 1A.2.34 indicates that under certain circumstances an affected line associated with a system may not be isolated from the secondary containment as part of corrective action. Explain under what circumstances this will be the case.

430.227e

Explain what the words “augmented Class D systems” mean in relation to the purchase of pressure boundary components of radioactive waste systems (See ABWR SSAR Subsection 1A.2.34) to assure their capability to provide integrity.

440.28

In SSAR Table 1.8-19, it is stated that branch technical position RSB 5-2 is applicable for ABWR. How does the ABWR design comply with BTP RSB 5-2?

20.2.2 Chapter 2 Questions

241.1

Table 2.01 in the Advanced BWR Standard Plant Safety Analysis Report (SSAR) gives an envelope of ABWR plant site design parameters. This table gives the minimum bearing capacity and the minimum shear wave velocity of the foundation soil. The table also gives the values of SSE and OBE and indicates (a) that the SSE response spectra will be anchored to Regulatory Guide (RG) 1.60, and (b) that the SSE time history will envelope SSE response spectra. The following additional information/clarification should be provided in the SSAR:

(1) While the SSE (PGA) of 0.3 g anchored to RG 1.60 could, in general, be considered conservative for many sites in the Central and Eastern United States, the SSAR should recognize and reflect the fact that localized exceedances if this value cannot be ruled out categorically and that adequate provisions will be made in the seismic design to consider site-specific geological and seismological factors.

(2) The SSAR gives an OBE (PGA) value of 0.10g and states that, "for conservatism, a value of 0.15 g is employed to evaluate structural and component responses in Chapter 3." The staff, however, considers the OBE value to be 0.15g as per criterion 2 of 10CFR50 Appendix A and paragraph V of 10CFR100 Appendix A which require, in part, that for seismic design considerations the OBE shall be no less than one-half of the SSE.

(3) The SSAR should indicate the procedures that would be adopted to evaluate the liquefaction potential at selected soil sites. It is not sufficient to say that the liquefaction potential will be "none at plant site resulting from OBE and SSE."

451.1

What are the bases (including references) for the site envelope of the ABWR design meteorological parameters listed in Table 2.0-1? Are these values intended to reflect the indicated maximum historical values for the contiguous USA? What is the combined winter precipitation load for the addition of the 100-year snow pack and the 48-hour probable maximum precipitation? What is the duration of the design temperature and wind speed values? What gust factors are associated with the extreme winds? Are any other meteorological factors (e.g., blowing dust) considered in the ABWR design?

451.2

Short-term dispersion estimates for accidental atmospheric releases are not provided explicitly in Section 2.4.3. If your X/Q values which are listed in Chapter 15 represent an upper bound for which the ABWR is designed; what is the bases for their selection?

20.2.3 Chapter 3 Questions

210.3

In Subsection 3.1.2.1.1.2, “Evaluation Against Criterion 1”, a footnote states that “important-to-safety” and “safety-related” are considered equivalent in this SSAR. The staff does not agree with this definition. The staff’s position on this issue remains as stated in NRC Generic Letter 84-01, “NRC Use of the Terms “Important to Safety” and “Safety-Related”, dated January 5, 1984. The staff used this position as guidance in its reviews of applications for operating licenses of nuclear power plants for a number of years prior to the issuance of GL 84-01. During these reviews, the staffs’ evaluations of the quality assurance requirements in 10CFR50, Appendix B generally applied to the narrower class of “Safety-related” equipment as defined in 10CFR50.49(b)(1). 10CFR100, Appendix A and in Section 3.2 of this SSAR. This implied that normal industry practice for quality assurance was generally acceptable for most equipment not covered by the “safety-related” definition. However, as pointed out in Generic Letter 84-01, there have been specific situations in the past where the staff has determined that quality assurance requirements beyond normal industry practice were needed for components and equipment in the more broad “important to safety” class.

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

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

210.4

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

210.5

In Table 3.2-1, Items B1.7, “Control Rods” and B1.9, “Fuel Assemblies” are classified as Safety Class 3, which is consistent with the criteria in the ANSI/ANS 52.1–1983 Standard. As stated in Question 210.4, the staff does not agree with all of the recommendations in that Standard. The staff position is that Control Rods and Fuel Assemblies should be Safety Class 2

and Quality Group B. To be consistent with this position and with staff reviews on recent BWR/6 plants, such as Perry and River Bend, revise Table 3.2-1 to change the classifications of the Control Rods and Fuel Assemblies from Safety Class 3 to 2 and add Quality Group B.

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

210.6

In Table 3.2.1, Item B2.5 identifies Main Steam Line (MSL) piping from the outermost isolation valve to and including the seismic interface restraint as being Safety Class 1 and Quality Group A. Figure 5.1-3b, “Nuclear Boiler system P&ID, Sheet 2” identifies the same portion of the MSL as Quality Group B. Beyond the seismic interface restraint, the MSL piping is quality Group D, which is not acceptable to the staff. To be acceptable, the MSL should be classified as recommended in Standard Review Plant 3.2.2, “System Quality Group Classification”, Appendix A, i.e., Quality Group B from the outermost isolation valve to the turbine stop valve. This staff position is based on the assumption that the ABWR MSL design differs from the BWR/6 design in that it does not contain a shutoff valve in addition to the two containment isolation valves. Revise Table 5.1-3b, Table 3.2-1, Subsection 3.9.3.1.3 and Subsection 5.4.9.3 to be consistent with the above staff position.

210.7

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

210.8

In Table 3.2-1, Item B3.1, the primary side recirculating motor cooling system piping is classified as Safety Class 3 and Quality Group C. In Subsection 3.9.3.1.4, this piping is described as being designed to the ASME Code, Section III, Subsection NB-3600, which is comparable to Safety Class 1. In Figure 5.4-4, “Reactor Recirculation System P&ID”, this piping is identified as Quality Group A. The staff's position is that this piping should be, as a minimum, Safety Class 1, Quality Group A and meet the requirements of 10CFR50, Appendix B from the interface of the piping with the pump motor casing to and including the first pipe

support. The remainder of this piping should be as a minimum, Safety Class 2. In addition, Item B3.2, the supports for this piping, should be the same Safety Class as the supported piping. Revise Items B3.1 and B3.2 in Table 3.2-1 to be consistent with the staff position.

210.9

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

210.10

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

210.11

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

210.12

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

210.13

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

210.14

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

210.15

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

210.16

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

G1.1—Vessels.

G1.2—Regenerative Heat Exchanges.

G1.3—Cleanup Recirculation Pump.

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

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

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

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

210.17

In Table 3.2-1, Items G2.3 “Heat Exchangers”, G2.4 “Pumps and Pump Motors”, G2.5, “Piping, Valves”, and G2.7 “RHR Connections” in the Fuel Pool Cooling and Cleanup System are all classified as Non-Nuclear Safety, which is consistent with the criteria in the ANSI/ANS 52.1-1983 Standard. As stated in Question 210.4, the staff does not agree with all of the recommendations in that Standard. The staff position is that all of the above items should be Safety Class 3, Seismic Category 1 and listed under quality Assurance requirements of 10CFR50, Appendix B. Regulatory Positions C.2 in Regulatory Guide 1.26 and C.1 in Regulatory Guide 1.29 includes this position. To be consistent with this position and with staff reviews on recent BWR/6 plants, such as Perry and River Bend, revise Table 3.2-1 to change the classification of Items G2.3, G2.4, G2.5, and G2.7 from Non-Nuclear Safety to Safety Class 3, add Seismic Category 1 and add “B” under Quality Assurance Requirement.

210.18

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

210.19

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

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

F4.1—Refueling equipment platform assembly.

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

The staff position, as discussed in Position C.1 and C.4 of Regulatory Guide 1.29 is that quality assurance requirements of 10CFR50, Appendix B should be applied to all structures, systems and components which are classified as Seismic Category 1. Revise Table 3.2-1 to add “B” in the Quality Assurance Requirement column for Item B3.1, F4.1, and F5.1.

210.20

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

210.21

In Subsection 5.2.1.1, Table 3.2-4 is reference to show the ABWR compliance with the rules of 10CFR50, codes and Standards. Subsection 3.2 in the SSAR does not contain a reference to Table 3.2-4. In either Subsection 3.2 or 5.2.1.1, provide the information requested in Standard Review Plant, Section 5.2.1.1, “Compliance With the Codes and Standards Rule, 10CFR50.55a”. This information should include the component Code, Code Edition and Code Addenda which will be applicable to ABWR pressure vessels, piping, pumps, valves, tanks, component supports and equipment.

210.22

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

210.23

Subsection 3.6.1.1.3(2) states that a pipe break event will not occur simultaneously with a seismic event. This does not agree with Standard Review Plant, Section 3.6.1, Branch Technical Position ASB 3-1, Paragraph B.2.b(1) or with the staff’s interpretation of Plant Event 8 in Table

3.9-2 of the SSAR. Revise Subsection 3.6.1.1.3(2) to be consistent with the staff position in SRP 3.6.1 or provide a justification for not doing so.

210.24

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

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

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

210.25

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

210.26

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

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

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

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

210.27

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

210.28

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

210.29

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

210.30

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

210.31

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

210.32

In Subsections 3.9.2.1.1 and 14.2.12.1, there is no mention of preoperational vibration testing of safety-related instrumentation lines. It is the staff's position that all essential safety-related

instrumentation lines and small borepiping should be included in the vibration monitoring program during preoperational or start-up testing. We require that either a visual or instrumented inspection (as appropriate) be conducted to identify any excessive vibration that could result in fatigue failure. Generally, this includes the portion up to and including the first support away from the connection to large bore piping or component. If observations suggest that other spans are being excited, further inspection would be conducted on a case by case basis. Revise the above Subsections to provide a commitment to this position.

210.33

The discussions in Subsection 3.9.2.5 and 3.9.5.2 relative to the dynamic system analysis of reactor internals under faulted conditions does not provide enough detailed information for the staff to evaluate. Standard Review Plan, Section 3.9.2.11.5 provides the acceptance criteria which the staff uses to evaluate this issue. Information in sufficient detail to implement this criteria is required before the staff can complete its evaluation. Revise Subsection 3.9.2.5 to include this information either in the form of references or an additional appendix in Section 3.2 of the ABWR SSAR.

210.34

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

210.35

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

210.36

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

210.37

Subsection 3.9.3.2 contains several references to IEEE-344, “IEEE Recommended Practices for Seismic Qualification of Class IE Equipment for Nuclear Power Generating Stations” with

no issue date. To be consistent with current staff positions on this issue, revise each of these references to read “IEEE STD. 344-1987” and add a commitment to NRC Regulatory Guide 1.100, Revision 2, “Seismic Qualification of Electrical Equipment in Nuclear Power Plants” to each reference. The staff considers these two documents to be applicable to mechanical as well as electrical equipment.

210.38

Subsection 3.9.3.3.2, “Other Safety-Relief Valves” references ASME Section III, Appendix 0 for the safety-relief valve opening and pipe reaction loads which will be used in the design of ABWR safety-relief valves. The staff's position on this issue is that if Appendix 0 is used, the additional criteria in Standard Review Plan, Section 3.9.3, Paragraph II.2 is applicable. Revise Subsection 3.9.3.3.2 to include a commitment to this position.

210.39

Subsections 3.9.3.4.1 and 3.9.3.5 both state that the jurisdictional boundary between component supports designed to ASME Section III, Subsection NF and the building structure shall be as defined in the project design specifications. The project design specifications may or may not agree with the definitions of jurisdictional boundaries which are in ASME Subsection NF. Therefore, revise Subsections 3.9.3.4.1 and 3.9.3.5 of the ABWR SSAR to provide a commitment that the 1987 Addenda to the 1986 Edition of ASME Section III, Subsection NF will be used to define the jurisdictional boundary between Subsection NF component supports and the building structure.

210.40

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

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

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

210.41

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

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

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

210.42

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

210.43

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

210.44

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

210.45

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

210.46

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

210.47

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

210.48

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

210.49

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

210.50

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

220.1

In section 3.5.3 for local damage prediction of concrete structures and barriers, the concrete wall and roof thicknesses determined should be less than those listed for Region II in Table 1 of SRP Section 3.5.3 unless justification is provided

220.2

The soil-structure interaction (SSI) analyses of the reactor building (RB) discussed in Section 3.7 of the ABWR SSAR are based on Revision 2 of SRP Sections 3.7.1 and 3.7.2 as provided for by the Licensing Review Bases dated August 7, 1987. It should be noted that Revision 2 is currently in the process of public comments and to this date has not been finalized. Consequently, there may be changes to Revision 2 which may require further discussion of this topic at a later date.

220.3

It is indicated that computer programs SASSI and CLASSI/ASD will be used to perform SSI analyses. Indicate how these programs are validated. In CLASSI the contribution of radiation damping cannot be determined on a mode by mode basis and it can have a substantial impact on building response. Provide results of sensitivity studies.

220.4

Since the response due to SSE are obtained in ratio to the response from the OBE analysis, indicate what is the purpose of establishing response spectra with.07 and 0.10 damping.

220.5

In Section 3.7.2.9, a number of conservative assumptions are listed in the calculation of floor response spectra. Some of the assumptions listed are not relevant to the generation of the floor response spectra, but to the overall design of the equipment. It is stated that the floor response spectra obtained from the time-history analysis of the building are broadened plus and minus 10% in frequency. In view of the fact that response spectra for all site-soil cases are combined to arrive at one set of final response spectra (Section 3.7.2.5), indicate how the ±10% broadening is accomplished.

220.6

In Subsection 3.7.3.2.2, for fatigue evaluation it is indicated that only 10 peak OBE stress cycles are taken into account which appears to be very low, considering the fact that the reactor building may also be subjected to SRV loadings. As indicated in the SRP Section 3.7.3 larger number of cycles should be considered.

220.7

In Appendix 3A.6 the following statement is made in the first paragraph:

> “The behavior of soil is nonlinear under seismic excitation. The soil nonlinearity can be conveniently separated into primary and secondary nonlinearities. The primary nonlinearity is associated with the state of deformations induced by the free-field ground motion. The secondary nonlinearity is attributed to the SSI effects. The secondary effect on structural response is usually not significant and is neglected in the appendix.”

Indicate if the secondary effect includes the radiation damping. If it does not, indicate how it is considered in the analysis.

220.8

In Appendix 3A.6 the computer program SHAKE is used to perform free-field site response analysis. To staff's knowledge, analysis based on SHAKE under certain site conditions may give unrealistic results and it cannot be used indiscriminately. In view of this observation, indicate what control or cause has been exerted in your use.

220.9
It is noted that ABWR is designed for 60-year life versus the 40-year life for plant design in current regulation. From the point of view of structures, provide your justification for the longer plant life.

220.10
Since the containment is integral with the reactor building, the following are staff's concerns:

(1) The thermal and pressure effects of the containment on the reactor building, especially under severe accident conditions.

(2) The restraint effects of the reactor building floor slabs on the behavior of the containment, especially on the ultimate capacity of the containment. (The staff has not received Chapter 19 which is believed to contain the estimate of the ultimate capacity).

(3) The behavior of small and large penetrations which span between containment and reactor building, especially under severe accident conditions.

Your approaches to resolve these concerns should be provided. If the resolution is to be accomplished through testing, provide a description of the tests to be performed.

220.11
In Subsection 3.8.4.3.1.2 it is noted that the main reinforcement in the containment wall consists of inside and outside layers of hoop and vertical reinforcement and radial bars for shear reinforcement. It appears that no diagonal seismic reinforcement is used. Indicate how the tangential shear due to horizontal earthquake is to be resisted.

220.12
In Subsection 3.8.4.3.1.2, for the same loads considered the first load combination under item (1), if compared with the first load combination under the (2), should obviously be the governing one. It appears that a re-examination of the load combinations in this section should be made to weed out load combinations which are obviously not controlling the design unless there are errors in the combinations. Furthermore since the RB is integral with the containment, effects due to such integration should be reflected in the load combinations of structural elements or components outside the containment unless considered otherwise.

220.13
The terms, G1, Gr and G all as defined in Subsection 3.8.1.3.1 are not listed in table 3.8.-1 while the terms 1v and ALL listed in Table 3.8.1 are not defined. Clarification of the table is requested.

220.14
In table 3.8-5 for load combination No. 3, it appears the acceptance criterion should be changed to S from U unless justified otherwise.

220.15

Discuss the potentials for severe accident that can be caused by external initiators such as high wind, tornado, tsunami, and earthquakes, and specifically flood since the reactor building has a standard soil embedment of 85 feet.

251.12

Criterion 51, “Fracture Prevention of Containment Pressure Boundary”, is only applicable for containments made of ferritic materials. Since the ABWR containment is made of concrete, this section should clarify the applicability of Criterion 51 to the ABWR containment. (3.1.2.5.2.1)

251.13

This section must include a discussion of all potential turbine missiles and mechanisms of missile generation. The turbine missile discussion should include failure of turbine discs and blades. (3.5.1.1.1.3)

251.14

This section must include a discussion of a favorable turbine orientation or provide a discussion on maintenance of the main steam turbine to protect against turbine missiles. (3.5.4.2)

251.15

Leak-Before-Break (LBB)—The staff considers LBB evaluations to be plant specific because parameters such as potential piping degradation mechanisms, piping geometry, materials, fabrication procedures, loads and leakage detection systems are plant specific. Therefore, the detailed LBB analysis should be provided when an application references the ABWR design (3.6.3)

271.1

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

271.2

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

410.1

Section 3.5.1, “Missile Selection and Description,” states: “The missile protection criteria to which the plant has been analyzed comply with the intent of 10CFR50 Appendix A, General Design Criteria for Nuclear Power Plants.” Provide a list of those instances where the protection criteria are in strict compliance with 10CFR50 Appendix A, and those instances where the protection criteria comply only with its “intent.” Provide an explanation of and justify the acceptability of those missile protection criteria which are in compliance only with “intent” of 10CFR50, Appendix A. (3.5.1)

410.2

Section 3.5.1 states: “A statistically significant missile is defined as one which could cause unacceptable plant consequences or violation of the guidelines of 10CFR100.” Provide an explanation of “unacceptable plant consequences.” (3.5.1)

410.3

Section 3.5.1.1, “Internally Generated Missiles (Outside Containment)” states: “Failure rates (P1) for value bonnets are in the range of 10^{-4} to 10^{-5} per year.” Provide a reference or analysis in support of the above statement. (3.5.1.1)

410.4

Regarding the physical separation requirements, provide a list of all systems (required for safe shutdown, accident prevention or mitigation of consequences of accidents) whose redundant trains do not have missile-proof barriers, and include the minimum separation distances. Provide, for the limiting case of the minimum separation distance, an analysis demonstrating the acceptability of the approach of not calculating P2, and instead relying on the “extremely low” probability of a missile strike to both trains, or a missile from one train striking the redundant train. (3.5.1.1)

410.5

Explain how safety-related systems or components are protected from missiles generated by non-safety-related components. It is the staff's position that missiles generated from nonsafety related components should not impact safety related components since a single active failure is assumed concurrent with the missile. (3.5.1.1)

410.6

Discuss the means by which stored spent fuel is protected from damage by internally generated missiles. (3.5.1.1)

410.7

Section 3.5.1.1.1.4, “Other Missile Analysis,” discusses the example of analysis of a containment high purge exhaust fan for a thrown blade. Provide the details of this analysis, such as the maximum penetration of the blade and the thickness of the fan casing. Discuss whether this analysis is conservative with respect to other rotating equipment missile sources. (3.5.1.1)

410.8

Regarding Subsection 3.5.1.1.2.2, “Missile Analysis,” provide the details of the rack, strap and cover assemble design for the pneumatic system air bottles, showing the thickness of the steel cover and the distance to the concrete slab. (3.5.1.1)

410.9

Regarding Subsection 3.5.1.1.3, “Missile Barriers and Loadings”, provide a list of all local shields and barriers outside intended to mitigate missile effects, giving their specific locations and design data. Provide an example of an analysis showing that the design of the shield or barrier will withstand the most energetic missile which could credible impact it. (3.5.1.1)

410.10

Section 3.5.1.2.1, “Rotating Equipment” (which can contribute to internally generated missiles inside the containment, states: “By an analysis similar to that in 3.5.1.1.1, it is concluded that no items of rotating equipment inside the containment have the capability of becoming potential missiles.” Provide the details of this analysis. (3.5.1.2)

410.11

Regarding Reactor Internal Pump (RIP) motors and impellers which can contribute to internally generated missiles inside the containment, explain the bases for concluding that the RIPs are incapable of achieving an overspeed condition and that the motors and impellers are incapable of escaping the casing and the reactor vessel wall (SSAR Subsection 3.5.1.2.1). Your response should explain how the provision of an anti-rotation device at the bottom of the RIP motor which prevents backward rotation of the RIP will prevent its overspeed during the course of a LOCA or during normal plant operation when one RIP is stopped and the other RIPs are operating (see SSAR Subsection 5.4.1.5). (3.5.1.2)

410.12

Regarding pressurized components, provide justification for the statement, “FMCRD mechanisms are not credible missile sources,” made in Subsection 3.5.1.2.2.

410.13

Regarding Subsection 3.5.1.2.3, “Missile Barriers and Loadings”*,* provide the same data for internally generated missiles inside the containment, as that requested under Question No. 410.8 above. (3.5.1.2)

410.14

Clarify whether secondary missiles generated as a result of the impact of primary missiles have been considered. Explain how protection against credible secondary missiles is provided. (3.5.1.2)

410.15

Regarding Subsection 3.5.1.2.3, “Evaluation of Potential Gravitational Missiles Inside Containment” Item 3, “Equipment for Maintenance,” describe any interface requirements imposed by this item on applicants referencing the ABWR. (3.5.1.2)

410.16

Regarding missiles generated by natural phenomena, provide the details of the tornado-missile analysis performed, identifying the tornado region (as defined in RG 1.76) and the missile spectrum. Discuss the compliance of the analysis with NUREG-0800, Subsection 3.5.1.4 acceptance criteria; Regulatory Guide 1.76, Positions C.1 and C.2; and Regulatory Guide 1.117, Positions C.1 through C.3 (3.5.1.4)

410.17

Provide specific descriptions of all provisions made to protect the charcoal delay tanks against externally generated tornado missiles. Discuss any interface requirement imposed by these design provisions.

410.18

Regarding SSC to be protected from externally generated missiles, discuss compliance with NUREG-0800, Subsection 3.5.2 acceptance criteria; Regulatory Guide 1.13, Position C2; Regulatory Guide 1.27, Positions C2 and C3; and Regulatory Guide 1.117, Positions C1 through C3. (3.5.2)

410.19

Clarify whether <u>all</u> nonsafety-related SSC, that may adversely impact (as a result of their failure due to an external missile) the intended safety function (i.e. achieving and maintaining safe shutdown, mitigating the consequences of an accident or preventing an accident) of a safety related SSC, are protected from external missiles. Describe how such SSC are protected. (3.5.2)

410.19a

SSAR Subsection 3.5.1.3.2.2, “Separation,” relies on physical separation between redundant essential systems including their related auxiliary systems as the basic protective measure against the dynamic effects of postulated pipe failures. The general arrangement drawings (e.g., Figure 1.2-2) are scheduled to be submitted in December 1988. Note that additional information on Subsection 3.6.1 may be requested as a result of the review of the above drawings. (3.6.1)

420.20

Section 3.6.1.1.1, “Criteria,” states that the overall design generally complies with BTP ASB 3-1. Specify those criteria which are in strict compliance, and those which are not in strict compliance with the BTP. Also, provide justification for the items that are not in strict compliance. (3.6.1)

410.21
Provide a listing of all the moderate-energy piping outside the containment, but within the scope of ABWR. Also, describe how safety-related systems are protected from jets, flooding and other adverse environmental effects that may result from pipe failures in moderate energy piping systems. (3.6.1)

410.22
Justify the non-inclusion of pipe failure analyses for the Process Sampling System, Fire Protection System, HVAC Emergency Cooling Water System and the Reactor Building Cooling Water Systems related to the Ultimate Heat Sink. Provide a summary table listing the protective measures provided against the effects of postulated pipe failures in each of the above systems and the systems listed in SSAR Tables 3.6-2 and 3.6-4. (3.6.1)

410.23
Give details for the worst case flooding arising from a postulated pipe failure and include the mitigation features provided. Note that for flooding analysis purposes, the complete failure of non-seismic Category I moderate-energy piping systems should be considered in lieu of cracks in determining the worst case flooding condition. (3.6.1)

410.24
Identify all the high-energy piping lines outside the containment (but within the ABWR scope), the adverse effects that may result from failures of applicable lines among them, and the protection provided against such effects for each of such lines (e.g., barriers and restraints). (3.6.1)

410.25
Clarify whether the reactor building steam tunnel is part of the break exclusion boundary. Also, provide a subcompartment analysis for the steam tunnel. Discuss how the structural integrity of the tunnel and the equipment in the tunnel are protected against failures in the tunnel. (3.6.1)

410.26
State how the MSIV functional capability is protected. (3.6.1)

410.27
Provide a summary table of the findings of an analysis of a postulated worst-case DBA rupture of a high or moderate-energy line for each of the following areas: 1) RCIC compartment, 2) RWCU equipment and valve room, 3) other applicable areas outside the containment (e.g., housing RHR piping). (3.6.1)

410.28
Clarify whether protection for safety-related systems and components against the dynamic effects of pipe failures include their enclosures in suitable design structures or compartments, drainage systems and equipment environmental qualification as required. If so, give typical examples for the above type of protection. (3.6.1)

410.29

Regarding interfaces (Section 3.6.4.1), include results of analyses of moderate-energy piping failures (currently, the interface requirements address only the high-energy piping failures analyses). (3.6.1)

410.29a

Appendix 3I, “Equipment Qualification Environmental Design Criteria,” is scheduled to be submitted in December 1988. Note that additional information may be requested based on review of the above appendix. (3.11)

410.30

Although there are no detailed qualification requirements for safety-related mechanical equipment in a harsh environment, GDC 1, “Quality Standards and Records,” GDC 4, “Environmental Missile Design Vases,” and Appendix B to 10CFR50, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants” (Section III, “Design Control,” and XVII, “Quality Assurance Records”) contain the following requirements related to equipment qualifications:

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

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

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

(4) Equipment qualification records shall be maintained and shall include the results of tests and material analyses.

Clarify whether the design complies with all the above requirements for safety-related mechanical equipment in a harsh environment within the ABWR scope. Provide justification for the non-compliance items above and identify any interface requirements needed to comply with the above. (3.11)

420.6

Identify the topical reports that will be provided to support any aspects of the design that are substantially different relative to designs previously reviewed by the staff. Subjects addressed in these topical reports should include but not necessarily be limited to the following:

> Methodology, basis and acceptance criteria for qualifying the system and equipment to the design basis electromagnetic interference (EMI) environment. (App 3I)

420.7

Identify the topical reports that will be provided to support any aspects of the design that are substantially different relative to designs previously reviewed by the staff. Subjects addressed in these topical reports should include but not necessarily be limited to the following:

Methodology, basis, and acceptance criteria for qualifying the system and equipment to the design basis surge withstand capability (SWC). (App 3I)

420.8

Identify the topical reports that will be provided to support any aspects of the design that are substantially different relative to designs previously reviewed by the staff. Subjects addressed in these topical reports should include but not necessarily be limited to the following:

> Methodology, basis, and acceptance criteria for qualifying the system and equipment to the design basis thermal environment established by localized heat transfer within electronic equipment, including in non-accident environments; this should also address requirements for humidity controls to preclude damage from electrostatic discharge. (App 3I)

420.9

Identify the topical reports that will be provided to support any aspects of the design that are substantially different relative to designs previously reviewed by the staff. Subjects addressed in these topical reports should include but not necessarily be limited to the following:

> Methodology, basis, and acceptance criteria for qualifying electronic and fiber-optic systems and equipment to the design basis radiation environment, including in environments normally considered “mild” for insulation materials. (App 3I)

420.84

What EMI coupling protection is to be provided for the I&C systems and how will its effectiveness for specific installed conditions be verified? (Examples of standards such as FCC docket 20780, Part 15, Subpart J, “Class A Computing Devices” have been identified by industry for computing devices as a source limitation for radiated and conducted noise. Also ANSI C63.12-1984 “Recommended Practice on procedures for Control of System Electromagnetic Capability,” is available as a design guidance tool.) Address these effects, possible limitations, and the criteria and standards to be used by GE in the ABWR design for safety systems equipment. (App 3I)

430.228

Criteria for the design basis for protection from external flooding should conform to Regulatory Guide 1.102, “Flood Protection for Nuclear Power Plants” as well as Regulatory Guide 1.59, “Design Basis Floods for Nuclear Power Plants”. Modify the statement in ABWR SSAR Subsection 3.4 to include the commitment to meet this Regulatory Guide. (3.4)

430.229
Flood protection analysis is provided for the reactor building and control building only. The ABWR SSAR scope includes structures, systems and components important to safety in this area. However, portions of other structures, within the scope of the plant-specific applicant may house systems and components important to safety (for example, the pumps associated with the ultimate heat sink). The SSAR therefore needs to specify as interface criteria flood protection design criteria for these systems, structures and components similar to those identified for internal and external flooding for the systems, components and structures within the ABWR SSAR scope. (3.4.1)

430.230
ABWR SSAR Subsection 3.4.1.1.1 references Figure 1.2-2 (which presumably includes a reference to Figure 1.2-2a). This section should also reference Figures 1.2-4 through 1.2-7 which provide a more complete view of safety-related components located below the design flood level. Additionally, these figures should be modified to show the location of all watertight doors used to provide compartment separation and the location of raised sills for which credit is taken. (3.4.1)

430.231
Section 3.4.1.1.2 references flooding from a feedwater line break in the steam tunnel, with data for the evaluation provided in Chapter 15.1. However, the evaluation is not provided in ABWR SSAR Subsection 3.4.1. Provide the flood analysis for this high energy line break. (3.4.1)

430.232
Your response to Question Nos. 430.73 and 430.85 (submittal dated February 28, 1990) states that the worst possible flood (circulating water system failure) that can affect the turbine building would result in a flood level slightly higher than grade and that all plant safety-related facilities are protected against site surface water intrusion (external flooding). Explain how all structures, systems and components (SSC) important to safety are protected against site surface water intrusion resulting from the above flood level. Also, considering access openings and penetrations below design flood level between the reactor building and turbine building (See ABWR SSAR Table 3.4-2), explain how the SSC important to safety located in the reactor building are protected from flooding inside the turbine building. (3.4.1)

430.233
Discuss how SSC important to safety are protected against flooding that may result from failure of non-safety-related plant equipment and components located outdoors (e.g., condensate storage tank). (3.4.1)

430.234
Identify the safety classification (seismic category, quality group) for all instrumentation used to alert the operator on flood situation for performing timely corrective actions. (3.4.1)

430.235

Provide flooding analyses for applicable plant areas to demonstrate that safety-related equipment and components of the fuel pool cooling and cleanup system and safety-related SSC in the fuel handling area will not be adversely affected by any postulated flooding; include flooding analysis for the radwaste and service buildings in so far as they relate to other structures which house SSC important to safety. Also, provide details to demonstrate that there is no uncontrolled leak path of radioactive liquid from the radwaste building under conditions of the worst-case internal flood. (3.4.1)

435.54

With regard to the classification of structures, components, and systems in Table 3.2-1; item R1 “DC Power Supply-Nuclear Island” and item R2 “Auxiliary AC Power System” are very general in their present form. We have therefore determined that Table 3.2-1, items R1 and R2, should be expanded to include the following list of items. Please incorporate these items into Table 3.2-1 adding any additional items necessary to make it a complete list.

R1 DC Power Supply-Nuclear Island

125 volt batteries, battery racks, battery chargers, and distribution equipment

Control and power cables (including underground cable system, cable splices, connectors and terminal blocks)

Conduit and cable trays and their supports

Protective relays and control panels

Containment electrical penetration assemblies

Motors

R2 Auxiliary AC Power System

6900 volt switchgear

480 volt load centers

480 volt motor control centers

120 VAC safety related distribution equipment, including inverters

Control and power cables (including underground cable system, cable splices, connectors and terminal blocks)

Conduit and cable trays and their supports*

Containment electrical penetration assemblies

Transformers

Motors

Load sequencers

Protective relays and control panels

Valve operators

* Raceway installations containing Class 1E cables and other raceway installations required to meet Seismic Category I requirements (those whose failure during a seismic event may result in damage to any Class 1E or other safety related system or components).

20.2.4 Chapter 4 Questions

252.1

Subsection 4.5.1.1 (1) should state: "The properties of the materials selected for the control rod drive mechanism must be equivalent to those given in Appendix I to Section III of the ASME Code, or parts A and B of Section II of the ASME Code, or are included in Regulatory Guide 1.85, except that cold-worked austenitic stainless steels should have a 0.2% offset yield strength no greater than 90,000 psi."

252.2

Subsection 4.5.1.1 (2) should state: "All materials for use in this system must be selected for their compatibility with the reactor coolant as described in Articles NB-2160 and NB-3120 of the ASME Code."

252.3

Subsection 4.5.2.2: The first sentence should read: "Core support structures are fabricated in accordance with the requirements of ASME Code, Section III, Subsection NG-4000, and the examination and acceptance criteria shown in NG-5000."

252.4

Subsection 4.5.2.3: The following statement should be added to the last sentence of the first paragraph: "The examination will satisfy the requirements of NG-5300."

252.5

Subsection 4.5.2.4 should state: "Furnace sensitized material should not be allowed."

252.6

Subsection 4.5.2.5 should state: "All materials used for reactor internals will be selected for their compatibility with the reactor coolant as shown in ASME Code Section III, NG-2160 and NG-3120. The fabrication and cleaning controls will preclude contamination of nickel-based alloys by chloride ions, fluoride ions, or lead."

430.1

Provide a failure modes and effects analysis of the control rod drive system (CRDS) in tabular form with supporting discussion to delineate the logic employed. The failure analysis should demonstrate that the CRDS can perform the intended functions with the loss of any active single component. These evaluations and assessments should establish that all essential elements of the CRDS are identified and provisions are made for isolation from nonessential CRDS elements. It should be established that all essential equipment is protected from common mode failures such as failure of moderate-and high-energy lines. The failure mode and effects analysis of the control rod drives should include water, air and electrical failures to CRDs and how the CRD system operation is affected due to air contamination or water contamination. Before finalizing the scope of the analysis, refer to ACRS subcommittee meeting proceedings on the ABWR dated June 1, 1988. It is noted that the above information is to be included in

Appendix 15B of the SSAR which will be submitted at a later date. However, the evaluation of the functional design of the reactivity control systems cannot be completed until this information is provided. (4.6)

440.1

SRP 4.6 identifies the following GDCs 23, 25, 26, 27, 28, and 29 in the acceptance criteria. Confirm that the reactivity system, described in Section 4.6 of the SSAR, meet the requirements of the above GDCs.

440.2

In Section 4.6.2.3.2.2 analysis of malfunction relating to rod withdrawal, it is stated, “There are known single malfunctions that cause the unplanned withdrawal of even a single control rod.” Confirm that this is a editorial mistake and correct it if so. Otherwise, explain in detail the basis for this statement and why this is acceptable.

440.3

In Section 4.6.1.2 it is stated that CRD system in conjunction with CRC&IS and RPS systems provides selected control rod run in (SCRRI) for reactor stability control. Describe in detail how SCRRI works.

440.4

In Figure 4.6-8a, CRD system P&ID, sheet 1, piping quality classes AA-D, FC-D, FD-D, FD-B, etc. are shown. Submit the document which explains these classes and relates them to ASME code classes.

440.5

In Figure 4.6-8b, the leak receiver tank is shown. What is the function of this tank? How big is this tank? Will a high level in the tank impact the operation of the control rod drive?

440.6

Identify the essential portions of the CRD system which are safety related. Confirm that the safety related portions are isolable from non-essential portions. (4.6)

440.7

In the old CRD system, the major function of the cooling water was to cool the drive mechanism and its seals to preclude damage resulting from long term exposure to reactor temperatures. What is the function of purge water flow to the drives? (4.6)

440.8

We understand that the LaSalle Unit 2 fine motion control rod drive demonstration test is still in progress. Submit the test results as soon as it is available.

440.9

In the present CRD system design, the ball check valve ensures rod insertion in the event the accumulator is not charged or the inlet scram value fails to open if the reactor pressure is above 600 psig. Confirm that this capability still exists in the ABWR design. (4.6)

440.10

In section 4.6.2.3.1, it is stated the scram time is adequate as shown by the transient analyses of Chapter 15. Specify the scram time. (4.6.2.3.2.1)

440.11

For both the low (“zero”) and operating power region describe the patterns of the control rod groups that are expected to be withdrawn simultaneously with the new rod system, and estimate the maximum for the total and differential reactivity worth of these groups. What sort of margin to period scram will exist in the low power range. (4.6)

440.12

Describe the relative core location of control rods sharing a scram accumulator. Can a failure of the scram accumulator fail to insert adjacent rods? If so, discuss the consequences of that failure. (4.6)

440.30

In SSAR Section 7.7.1.2, Section 4, it is stated that: “The Rod Control and Information System (RC&IS) is not classified as a safety related system, it has a control design basis only and is not required for the safety and orderly shutdown of the plant. A failure of RC&IS will not result in fuel damage. The Rod block Functions of the RC&IS, however, are important in limiting the consequences of a rod withdrawal error during normal plant operation. An abnormal operating transient that might result in local fuel damage is prevented by the rod block enforcement functions of the RC&IS.”

If credit for RC&IS is assumed in the analysis of the rod withdrawal transient to meet the GDC 10 requirement that “specified acceptance fuel design limits (SAFDL) will not be exceeded,” the staff requires that RC&IS satisfies GDC-1 which states that “structures, systems and components important to safety must be designed, fabricated, erected and tested to quality standards commensurate with the safety function to be performed.”

440.31

Selected control rod run in (SCRRI) is provided for thermal-hydraulic stability control. Describe in detail how SCRRI controls stability.

440.32

We understand that the control rod has no velocity limiter. Discuss in detail the reason for velocity limiter elimination.

440.33

In SSAR Section 7.7.1.3, Section 7, 3 trips are described in the RPT logic. Do these 3 trips include the ATWS RPT trip or is the ATWS RPT trip separate?

100.1

In light of the recent interest in BWR thermal hydraulic stability following the LaSalle instability event, it appears to be highly desirable to assure that in an advanced BWR design the possibility of instability is precluded, both in normal and anticipated abnormal operating conditions; this should be the case without requiring the prompt intervention of the operator. If actions are required, they should be automatic. If operator attention is required, suitable monitoring capability should be readily available. Please discuss the extent to which this is provided for in the ABWR. This discussion should consider (1) the various potential problem areas which have been identified in the current BWR stability review (particularly asymmetric oscillations), (2) the relevant stability related characteristics of the ABWR core such as fuel entrance loss coefficients, void reactivity coefficients, fuel conductivity, and including extremes of conditions in both the initial core and potential reload cores with different fuel, (3) accessible stability significant regions of the power-flow map, involving both normal and abnormal events (including multiple out of service or tripped recirculation pumps), (4) the Selected Control Rod Run-In (SCRRI), describing its relevant characteristics including provisions for automatic initiation, speed of operation compared to need for rapid action, boundaries of operation (on power-flow map), flexibility of these boundaries as need for change may arise, (5) the existing relevant instrumentation and the possible need for improved or augmented instrumentation such as on line stability measurement or easily available relevant LPRM readings and automatic action based on these measurements, (6) the need for frequent mapping of boundaries of operational map regions to be avoided. (Chapter 4)

20.2.5 Chapter 5 Questions

210.1

In Subsection 5.2.1.2, the statement is made that Section 50.55a of 10CFR50 requires NRC staff approval of ASME code cases only for Class 1 components. Revise this statement to be consistent with the current (1987) edition of 10CFR50.55a, which requires staff approval of code cases for ASME Class 1, 2, and 3 components.

210.2

Revise Table 5.2-1 or provide additional tables in Subsection 5.2.1.2 which identify all ASME code cases that will be used in the construction and in-plant operation of all ASME Class 1, 2, and 3 components in the ABWR. All code cases in these tables should be identified by code case number, revision, and title. These tables should include those applicable code cases that are listed either as acceptable or conditionally acceptable in Regulatory Guides 1.84, 1.85, and 1.147. For those code cases listed as conditionally acceptable, verify that the construction of all applicable components will be in compliance with the additional Regulatory Guide conditions.

250.1

Subsection 5.2.4.1 should state that the system boundary includes all pressure vessels, piping, pumps, and valves which are part of the reactor coolant system, or connected to the reactor systems, up to and including:

(1) The outermost containment isolation valve in system piping that penetrates the primary reactor containment.

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

(3) The reactor coolant system and relief valves.

250.2

Subsection 5.2.4.2 should satisfy the requirements in ASME Code IWA-1500.

251.1

Subsection 5.3.1.1 should state that the materials will comply with the provisions of the ASME Code, Section III, Appendix I, and meet the specification requirements of 10CFR50, Appendix G.

251.2

Subsection 5.3.1.2 should state the specific subsection NB of ASME Code to which the manufacturing and fabrication specifications were alluded.

251.3

Subsections 5.3.1.4.4 and 5.3.1.4.5 should be rewritten; the cross-reference is unacceptable.

Subsections 5.3.1.4.7, 5.3.1.5.2, 5.3.1.6.3, and 5.3.2.1.5: Revision 2 of Regulatory Guide 1.99 should be added in these subsections.

251.4

Subsection 5.3.1.6.1: the third capsule of the vessel surveillance program is designated as a standby; however, according to ASTM 185-82, the capsule should be withdrawn at the end of life. Provide justification for this deviation.

251.5

Subsection 5.3.1.6.3 states that according to estimates of worst-case irradiation effects, the adjusted reference temperature at end-of-life is less than 100°F, and the end-of-life upper shelf energy exceeds 50 ft-lb. Provide the calculation and analysis associated with the estimate.

251.6

Subsection 5.3.2.1 should clarify where Reference 2 is located. Has the NRC staff reviewed and approved Reference 2? If not, the staff needs to review Reference 2 in order to complete the review of this subsection.

251.7

Subsections 5.3.2.1.1, 5.3.2.1.2, 5.3.2.1.3, and 5.3.2.1.5 need to be rewritten. The level of detail must be comparable to that of Standard Review Plan 5.3.2 and Branch Technical Position MTEB 5-2.

252.8

Subsection 5.3.3 cited three GE documents:

(1) GE quality assurance program

(2) “Approved” inspection procedures, and

(3) NEDO-10029.

Has the NRC staff reviewed and approved the above documents? The staff cannot satisfactorily review this subsection without reviewing the above three documents.

251.9

Subsection 5.3.3.1.1.1 discusses the 60-year life of the ABWR reactor vessel. The NRC requirements and calculations on the fracture toughness and material properties are based on a 40-year life. Provide justification for the applicability of NRC’s requirements on the 60-year life reactor vessel.

251.10

Subsection 5.3.3.2 should include the following information: neutron fluence, shift in reference temperature RT_{NDT} and upper shelf energy. The staff needs this information to compare to that of predicated values using Regulatory Guide 1.99.

251.11

Subsection 5.3.3.6 should indicate that operating conditions should satisfy the pressure-temperature limits prescribed in Subsection 5.3.2.

252.7

Subsection 5.2.3.2.2 is mostly an academic discussion of BWR water chemistry effect on intergranular stress corrosion cracking (IGSCC) in sensitized stainless steels. The subsection should discuss the actual ABWR water chemistry effects on the IGSCC. The subsection is vague about specific remedies or preventive measures to avoid IGSCC in ABWR. For example, the subsection failed to discuss how much hydrogen is needed for injection into the feedwater system or how the “tight conductivity control” would be implemented.

Also provide references for the “Laboratory studies...” and “available evidence...” that were mentioned in this subsection.

252.8

Subsection 5.2.3.2.3 should state that the requirements of GDC 4, relative to the compatibility of components with environmental conditions are met by compliance with the applicable provisions of the ASME Code and by compliance with the recommendation of Regulatory Guide 1.44.

Specify the “very low limits” of the contaminants in the reactor coolant.

252.9

Subsection 5.2.3.3.1 should clarify where and how was the 45 ft-lb Charpy V value obtained.

The ferritic material used for piping, pumps, and valves should comply with Appendix G, Section G-3100, of ASME Code Section III.

This subsection should indicate that “calibration of instruments and equipment shall meet the requirements of the code, Section III, Paragraph NB-2360.”

252.10

Subsection 5.2.3.4.1.1 should be rewritten to include more detailed discussion on avoidance of significant sensitization and on how the ABWR design complies with the NRC regulatory requirements.

252.11

Subsection 5.2.3.4.2.3 states that the ABWR design meets the intent of this Regulatory Guide (1.71) by utilizing the alternate approach given in Section 1.8. We cannot review this subsection because we have not received Section 1.8. In addition, this subsection should be rewritten because it lacks detailed discussion about welder qualification.

281.1

In Section 5.1 (page 5.1-2) the function of the reactor cleanup system filter demineralizer should include the removal of radioactive corrosion and fission products in addition to particulate and dissolved impurities.

281.2

In Subsection 5.2.3.2.2 (page 5.2-7) irradiation-assisted stress corrosion cracking (IASCC) of reactor internal components and its mitigation are not discussed. Present laboratory data and plant experience has shown that IASCC can be initiated even at low conductivity (< 0.3µS/cm) after long exposure to radiation.

281.3

In Subsection 5.2.3.2.2 (pages 5.2-7 and 8) the ABWR Standard Plant design does not clearly incorporate hydrogen water chemistry to mitigate IGSCC. Since the plant design life is 60 years, hydrogen water chemistry may be of greater importance in reducing reactor coolant electrochemical corrosion potential to prevent IGSCC as well as IASCC. If hydrogen water chemistry is the referenced ABWR standard design, the following documents should be cited:

EPRI NP-5283-SR-A, "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations"—1987 Revision.

EPRI NP-4947-SR-LD, "BWR Hydrogen Water Chemistry Guidelines"—1987 Revision (to be published).

281.4

In Subsection 5.2.3.2.2 (page 5.2-9) the utilization of the General Electric zinc injection passivation (GEZIP) process for radiation buildup control for the ABWR is not discussed. GEZIP was identified as a required design feature in the ABWR presentation to NRC staff.

281.5

In Subsection 5.2.3.2.2 (page 5.2-9) prefilming of stainless steel appears to be a promising method to reduce the buildup rate of activated corrosion products during subsequent plant operation. SIL No. 428 recommends preoperational testing of the recirculation system conducted at temperatures 230°F be done with the dissolved oxygen level controlled to between 200 and 400 ppb. Is control of radiation buildup through preoperational oxygen control being considered for the BWR Standard Plant? Are mechanical polishing and electropolishing of piping internal surfaces also being considered for reducing radiation buildup?

281.6

In Subsection 5.2.3.2.2.2 (page 5.2-9) cobalt 60 is identified as the principle contributor to shutdown radiation levels, especially the recirculation piping system of BWRs. Stellite contributes about 90% of the total cobalt 59 input to the reactor water (EPRI NP-2263, BWR Cobalt Source Identification, February 1982). Since irradiation of cobalt 59 yields cobalt 60, reduction in the source of cobalt 59 is needed to reduce the buildup of shutdown radiation

levels. Indicate Stellite surface areas (square feet) in nuclear steam supply system and balance of plant. Provide the criteria for selecting Stellite plant materials for the designed application. Provide evaluation of noncobalt-containing materials whose properties are adequate to replace Stellite in-plant applications.

281.7

Subsection 5.2.3.2.2.3(4) (page 5.2-10) states that control of reactor water oxygen during startup/hot standby may be accomplished by utilizing the de-aeration capabilities of the condenser. In addition, this section states that independent control of control rod drive (CRD) cooling water oxygen concentrations of < 50 ppb during power operation is desirable to protect against IGSCC of CRD materials. Are either one or both of the above dissolved oxygen controls incorporated in the ABWR Standard Plant design?

281.8

In Subsection 5.2.3.2.2.3(13) (page 5.2-11) it states that the main steam line radiation monitor indicates an excessive amount of hydrogen being injected. An explanation of this occurrence should be discussed.

281.10

In the October 1987 ABWR presentation to the NRC staff the design features and/or requirements to improve water chemistry for GE-ABWR were specified. Address each one of these design features and/or requirements listed in Table I in the ABWR Standard Safety Analysis Report.

Table 20.2-1 Comparison of requirements in ABWR standard safety analyses report and ABWR presentation to NRC staff (October 21 and 22, 1987)

		ABWR Presentation to NRC Staff	ABWR Standard Safety Analysis Report
1 -	Selection of low cobalt materials to minimize radiation buildup	Required Design Feature	Not discussed in Subsection 5.2.3.
2 -	Hydrogen water chemistry to suppress IGSCC	Required Design Feature	Subsection 5.2.3.2.2 references normal water chemistry
3 -	Zinc injection to minimize radiation buildup	Required Design Feature	Not discussed in Subsection 5.2.3.2.2.2.
4 -	Full flow deep bed condensate system to reduce feedwater impurities	Required Design Feature	Not discussed in Subsection 5.2.3.2.2.3.
5 -	Improved online monitoring instrumentation to assure water quality	Ion chromatography, electrochemical corrosion potential, and crack arrest verification system required design features	Only electrochemical corrosion potential discussed in Subsection 5.2.3.2.2.3.
6 -	Improved corrosion-resistant materials for steam extraction piping to minimize feedwater impurities	Required Design Feature	Not discussed in Subsection 5.2.3.2.2.3.
7 -	Highly corrosion-resistant condenser tubes to minimize leakage into condensate system	Required Design Feature	Not discussed in Subsection 5.2.3.2.2.3.
8 -	Maintain electrochemical corrosion potential <0.23 V to suppress IGSCC	Required Design Feature	Not listed in Table 5.2-5.
9 -	Erosion/corrosion-resistant materials in steam extraction and drain lines to minimize failures	Design Feature	Not discussed in Subsection 5.4.9.
10 -	Ease of lead detection in and repair of the main condenser	Design Feature	May be in Subsection 10.4.1 which has not been submitted yet.
11 -	2% Reactor water cleanup system to improve water quality and occupational radiation exposure	Design Feature	Not discussed in Subsection 5.2.3.2.2.
12 -	Full flow recirculation to main condenser from cleanup output to reduce feedwater impurities.	Design Feature	Not discussed in Subsection 5.2.3.2.2.3.

430.2

Regarding Reactor Coolant Pressure Boundary (RCPB) leakage detection systems, provide information on the following: (5.2.5)

(1) Describe how the leakage through both the inner and outer vessel head flange seals will be detected and quantified.

(2) List the sources that may contribute to the identified leakage collected in the Reactor Building Equipment Drain Sumps.

(3) Describe how potential intersystem leakages will be monitored for the 1) Low Pressure Coolant Injection System, 2) High Pressure Core Spray System, 3) Reactor Core Isolation Cooling System (RCIC)-Water side and 4) Residual Heat Removal System Inlet and discharge sides. Your response should include **all** the applicable (for the ABWR design) systems and components connected to the Reactor Coolant System that are listed in Table 1 of SRP Section 5.2.5 and other systems that are unique to ABWR (except those that you have already discussed in SSAR Subsection 5.2.5.2.2, Item 11).

430.3

Discuss compliance of reactor coolant leak detection systems with Regulatory Guide (RG) 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", Positions C4, C5, C6, C8, and C9 with respect to the following items: (5.2.5)

(1) Indicators for abnormal water levels or flows in all the affected areas in the event of intersystem leakages.

(2) Sensitivity and response time of leak detection systems used for unidentified leakages outside the drywell.

(3) Qualification relating to seismic events for drywell equipment drain sump monitoring system and leak detection systems outside the drywell.

(4) Testing Procedures-Monitoring sump levels and comparing them with applicable flow rates of fluids in the sumps.

(5) Inclusion of reactor building and other areas floor and equipment drain sumps in ABWR Technical Specifications for leak detection systems.

Note that a few of the questions above arise because in Subsection 5.2.5.4.1 you state that the total leakage rate includes leakages collected in drywell, reactor building and other area floor drain an equipment drain sumps.

430.4
Clarify whether the RCIC makeup capacity is sufficient to provide also for main turbine stop valves. Also, clarify whether this leakage is included in the total leakage mentioned in Subsection 5.2.5.4.1.

430.5
Clarify how Position C.2 of RG 1.29, "Seismic Design Classification" is met for all applicable leak detection systems (also include the leak detection systems outside the drywell). (5.2.5)

430.6
Identify all the interface requirements relating to RCPB leakage detection systems. (5.2.5)

440.13
ODYNA and REDYA are the improved versions of NRC approved ODYN and REDY Codes. Describe the changes made in the codes. The staff requires approval of these codes before the final design approval.

440.14
Information given in NEDE-24011-P-A is not sufficient to demonstrate compliance with the ASME code. The ASME Code Section III, Article NB-7200, requires that an overpressure protection report be prepared. Provide this report for the staff review.

Include the following items in the report:

(1) Provide all system and core parameter initial values assumed in the overpressure analyses. Include their nominal operating range with uncertainties and Technical Specification limits.

(2) Scram time characteristics.

(3) Safety/relief valve characteristics.

(4) Demonstrate available safety margin considering the most limiting transients.

(5) Peak vessel bottom pressure versus time for the limiting transients.

(6) Provide graphical representation for peak vessel bottom pressure versus safety/relief valve capacity and number of safety/relief valves used for the most limiting transient.

(7) Identify conservatisms used in the overpressure transient analyses.

440.15
Confirm that the overpressure analysis includes the effects of the ATWS reactor recirculation pump trip on high reactor pressure.

440.16

Provide the sensitivity study which shows that increasing the initial operating pressure (up to the maximum permitted by the high pressure trip setpoint) will have a negligible effect on the peak transient pressure.

440.17

The performance of essentially all types of safety/relief valves has been less than expected for a safety component. Because of reportable events involving malfunctions of these valves on operating BWRs, the staff is of the opinion that significantly better safety/relief valves performance should be required of new plants. Provide a detailed description of improvements between your plant and presently operating plants in the areas listed below. In addition, explain why the noted difference will provide the required performance improvement.

(1) “Weeping” of SRVs is a generic problem. The following table explains the seriousness of the problem.

Comparison Of BWR/6s “Weeping” SRVs

Plant	**“Weeping” SRVs/Total No.**
Clinton	3/16
River Bend	12/19
Grand Gulf	11/20
Grand Gulf (after all valves changed during 1st refueling	6/20
Perry	18/19

The continuous “weeping” of the SRV has the potential to degrade SRVs and increase the frequency of use of RHR heat exchangers.

How will the ABWR SRVs resolve the generic problem stated above?

(2) **Valve and valve operator type and/or design**. Include discussion of improvements in the air actuator, especially materials used for components such as diaphragms and seals. Discuss the safety margins and confidence levels associated with the air accumulator design. Discuss the capability of the operator to detect low pressure in the accumulator(s). Provide detailed description of safety and relief mode of operation/function of the SRV.

(3) **Specifications**. What new provisions have been employed to ensure that valve and valve actuator specifications include design requirements for operation under expected environmental conditions (esp. temperature, humidity, and vibration)?

(4) **Testing**. Prior to installation, safety/relief valves should be proof tested under environmental conditions and for time period representative of the most severe operating conditions to which they may be subjected.

(5) **Quality Assurance**. What new programs have been instituted to assure that valves are manufactured to specifications and will operate to specifications.

(6) **Valve Operability**. Provide a summary of the surveillance program to be used to monitor the performance of the safety/relief valves. Identify the information that will be obtained and how these data will be utilized to improve the operability of the valves.

(7) **Valve Inspection and Overhaul**. Operating experience has shown that safety/relief valve failure may be caused by exceeding the manufacturer's recommended service life for the internals of the safety/relief valve or air actuator. At what frequency do you intend to visually inspect and overhaul the safety/relief valve? For both safety/relief and ADS modes, what provisions exist to ensure that valve inspection and overhaul are in accordance with the manufacturer's recommendations and that the design service life would not be exceeded for any component of the safety/relief valve?

440.18

Address the following TMI-2 action items related to SRVs.

(1) II.K.3.16

(2) II.B.1

(3) II.D.3

(4) I.K.3.28

(5) II.D.1

440.19

Explain in detail how the spring and relief modes of the SRV works. Are they an difference from the SRVs currently used in operating BWRs?

440.20

What ATWS considerations have you given for sizing SRVs?

440.21

In Subsection 5.2.2.2.2.3, the reclosure pressure setpoint (% of operating setpoint) for both modes are given as 98 and 93. Explain the significance of these numbers.

440.22

In Figure 5.1-3a the SRV solenoid valves are not shown as DC powered as they should be. Note 8 states that “valve motor operators and pilot solenoids are AC operated unless otherwise specified.”

440.24
Confirm that SRVs are designed to meet seismic and quality standards consistent with the recommendations of Regulatory Guides 1.26 and 1.29.

440.28
In SSAR Table 1.8-19, it is stated that branch technical position RSB 5-2 is applicable for ABWR. How does the ABWR design comply with BTP RSB 5-2?

440.29
Describe the methods planned for performing hydrostatic tests on ABWR RPV vessel after the initial start-up. Can you perform hydrostatic tests and leak tests without using critical heat?

440.34
In SSAR Chapter 5.4.1.4, it is stated “During various moderately frequent transient, various Reactor Internal Pump (RIP) operating modes will be required such as: Bank of five RIPs runback to 30% speed; trip from current speed conditions; or runback to 30% speed and subsequent trip. These control actions are all produced through control actions of the Recirculation Flow Control System (RFCS).”

Even though credit is taken for RFCS to mitigate transients as above, RFCS is not classified as a safety grade system. (See SSAR Chapter 7.7.1.3, Section 2).The staff has the same concern as given in Question No. 440.30.

440.35
In SSAR Chapter 5.4.1.5, it is stated “The recirculation system has sufficient flow coastdown characteristics to maintain fuel thermal margins during an abnormal operation transient.” What are the coastdown characteristics? Explain in detail why they are sufficient.

440.36
RCIC is taken credit in the LOCA analysis. What “upgrade” has been made to the ABWR RCIC system which is different from the BWR/6 RCIC system?

440.37
Traditionally, RCIC can be started with only reactor steam and DC power and it is independent of AC power for start up. Is this true for the ABWR RCIC?

440.38
In SSAR Chapter 5.4.6.1, Section 5, it is stated that “should a complete loss of AC power occur, RCIC is designed to operate for at least 30 minutes.” Typically, in current operating BWRs, the batteries (DC power) are available for at least 4 hours after station blackout. If the batteries are available for at least 4 hours, why is RCIC designed to operated for only 30 minutes?

440.39
Some of the recent BWRs licensed to operate have gland seal compressor instead of the gland seal condenser. Why the switch now to gland seal condenser? Is the ABWR gland seal

condenser design the same as the old design? Describe in detail the operation of condensate and vacuum pumps.

440.40
In SSAR Chapter 5.4.6.2.1.3, Section 2, it is stated that “the F031 limit switch activates when fully open and closes F022 and F059.” This interlock is not applicable when the system is in the test mode. During test mode the RCIC pump takes suction from the suppression pool and returns to the pool. Therefore, all the 3 valves will be open simultaneously. Correct the interlock description for F031, F022 and F059.

440.41
In the ABWR design, RCIC is tested by taking suction from the pool and turning to the pool. This new testing, unlike current plants where RCIC is tested from the condensate storage tank (CST), is a requirement to take credit as an ECCS system. But from an operational point of view, it is better to provide the test flow path from CST and to CST also. Normally, suppression pool water is a low quality water and hence, draining, flushing and filling of the system is required before putting the system back on standby after testing. (Normally, the system is lined up from CST). This may add unnecessary radiation exposure to operations personnel. We suggest that you consider adding a test return line to CST also. Since a suction line form CST is already provided, addition of new test return line to CST at the pump discharge should not be major change.

440.42
Why are the power supply for valves F063, F064, F076, F077, and F078 standby AC instead of DC?

440.43
Address the following TMI-2 action Items related to RCIC.

(1) II.K.1.22

(2) II.K.3.13

(3) II.K.3.15

(4) II.K.3.22

(5) II.K.3.24

440.44
Confirm that the RCIC system meets the guidelines of Regulatory Guide 1.1 regarding pump Net Positive Suction Head (NPSH).

440.45
SRP 5.4.6 identifies GDCs 5, 29, 33, 34 and 54 in the acceptance criteria. Confirm that the RCIC system, described in Chapter 5.4.6 of the SSAR, meets the requirements of the above GDCs.

440.46
In SSAR Chapter 5.4.6.3, it is stated “The analytical methods and assumptions in evaluating the RCIC system are presented in Chapter 15 and Appendix 15A.” Identify the section in Chapter 15 where the analytical methods and assumptions evaluation the RCIC systems are given.

440.47
Normally the RCIC pump takes suction from the condensate storage tank (CST). But the CST is not seismically qualified or safety related. Confirm that the system piping and level transmitters, which interface with CST, will be designed and installed such that the automatic switchover to the suppression pool takes place without failure.

440.48
The equipment and component description given in 5.4.6.2.2 is very brief. What type of turbine is used in the ABWR? Is it the same type as the Terry Turbines used in current BWRs? Is the turbine testing done by Terry Co. with water applicable to the ABWR? Describe in detail the components, especially the turbine and the pump.

440.49
To the best of our knowledge, the steam isolation valves F063 and F064 in currently operating BWRs are not tested with a steam pipe break downstream and with actual operating conditions (pressure 1000 psig and temperature 546°F). There is no guarantee that the steam isolation valves will close during a break. We require that a proper testing of the valves be performed before the final design approval. (Reference Generic Issue GI-87 “Failure of HPCI Steam Line Without Isolation.”)

440.50
Steam isolation valves F063 and F064 are to be opened in sequence to reduce water hammer and for slow warm-up of the piping. F064 and F076 are opened first. The valves logic should prevent the operator from opening the valves out of sequence. Confirm that the valves control logic includes an interlock.

440.51
Describe how the system design reduces water hammer. Confirm that a condensing sparger will be provided at the turbine exhaust to reduce water hammer. Add a necessary not in the P&I-D to indicate that the steam supply and exhaust lines are to be sloped to reduce water hammer.

440.52
The RCIC operation from the suppression pool may be limited by an increase in suppression pool water due to lube oil cooling done by suppression pool water. What is the maximum suppression pool temperature at which RCIC can be operated safely.

440.53
How is thermal shock prevented at the feedwater line injection point?

440.54
What is the minimum quantity of water required in the condensate storage tank (CST) for RCIC operation? Give the basis for the required quantity of water in the CST.

440.55
In the LOCA analysis (SSAR Table 6.3) 800 gpm is taken credit for the RCIC system. Due to pump degradation and flow controller measurement inaccuracies, the system may not deliver 800 gpm. The required system flow should be increased, accounting for uncertainties, to meet the LOCA analysis required flow.

440.56
In SSAR Chapter 5.4.6.2.1.3, Section 1, it is stated "there are two key-locked valves (F068 and F069) and two key-locked isolation resets." Change the description to state that the valves F068 and F069 are key-locked open.

440.57
What is the closing time of test return valves F022 and F059? They should close earlier than 15 seconds to prevent any flow diversion to the suppression pool during a LOCA.

440.58
Since RCIC is part of the ECCS network, the RCIC pump minimum flow line should be designed to operate for a reasonable length of time. How long can RCIC run in minimum flow mode?

440.59
What is the difference between Low Pressure Flooder System and Low Pressure Core Spray System? Describe in detail why "flooder" system is better than core spray. Submit detailed drawing showing the "flooder" inside the vessel.

440.60
In SSAR Table 1.3.2, it is stated that the RHR heat exchanger duty for suppression pool cooling is based on assuming they are placed in operation 20 hours after reactor shutdown.

This statement is not consistent with the normal assumption that suppression pool cooling is stated within ten minutes after a LOCA. What is the basis for sizing the RHR Hx? In SSAR Chapter 5.4.7.3.2, it is stated that ATWS was considered for RHR heat exchanger sizing. But a

Feedwater Line Break (FWLB) is the most limiting event. Describe in detail why FWLB is the limiting event and not ATWS.

440.61

In SSAR Chapter 5.4.7.3.2, Section 2, it is stated “because it takes 4 to 6 hours to reach the peak pool temperature, shutdown cooling will be initiated before peak pool temperature. The energy release from the reactor will be controlled by the shutdown cooling system, and there is no need to release the reactor energy to the pool.”

Which scenarios are postulated for the assumption stated above? For most scenarios, suppression pool cooling is started within a short time. Shutdown cooling is started at a much later stage. Describe in detail the assumptions made for sizing the RHR heat exchangers.

440.62

SRP 5.4.7 identifies GDCs 2, 5, 19 and 34 in the acceptance criteria. Confirm that the RHR system, described in Chapter 5.4.7 of the SSAR, meets the requirements of the above GDCs.

440.63

Confirm that the RHR system satisfies the requirements of TMI-2 Action item III.D.1.1.

440.64

Confirm that the RHR system meets the guidelines of Regulatory Guide 1.1 regarding pump Net Positive Suction Head.

440.65

In Section 5.4.7.2.3.1 (3) it is stated that “redundant interlocks prevent opening the shutdown connections to and from the vessel whenever the pressure is above the shutdown range.”

RSB 5-1 requires that the suction and discharge valves interfacing with the RCS shall have independent <u>diverse</u> interlocks to prevent the valves from being opened unless the RCS pressure is below the RHR design pressure.

Confirm that the high/low pressure interface with RCS satisfies the requirements of RSB 5-1.

440.72

NRC Bulletin 88-04 dated May 5, 1988, discusses the potential safety related pump loss. The first concern involves the potential for the dead-heading of one or more pumps in safety related systems that have a miniflow line common to two or more pumps or other configurations that do no preclude pump-to-pump interaction during miniflow operation. A second concern is whether or not the installed miniflow capacity is adequate for even a single pump in operation.

In the ABWR design, HPCS pump miniflow lines and test return lines to the suppression pool are routed through the RHR “c” loop test and minimum flow lines. How does the ABWR design satisfy the concerns given in NRC Bulletin No. 88-04?

440.73

In RHR process diagrams 5.4-11b, RHR heat exchanger removal capacity for different modes is not given. Revise the process diagram to include the heat removal capacity.

440.74

In Figure 5.4-10b, (I-12) flammability system (T-49) is cross-tied to the RHR system. What is the purpose of this cross-tie to the RHR system?

20.2.6 Chapter 6 Questions

250.3

Subsection 6.6.8 should discuss the augmented inservice inspection for those portions of high energy piping enclosed in guard pipes.

252.12

Subsection 6.1.1.1 should discuss ferritic steel welding in detail. It should also discuss the control of ferrite content in stainless steel weld metal similar to that of Regulatory Guide 1.31.

252.13

Subsections 6.1.1.1.3.1, 6.1.1.1.3.2, and 6.1.1.1.3.5 should be rewritten because the cross-reference is unacceptable.

281.9

Subsection 6.4.4.2 (page 6.4-6) discusses personnel respirator use in the event of toxic gas intrusion into the control room. However, the chlorine detection system is not discussed. Also, any control functions that are automatically triggered by a chlorine detector alarm (closing intake dampers, energizing control room HVAC system recirculation) should be identified.

430.7

In the SSAR section devoted to containment functional design, identify clearly those areas that are not part of the ABWR scope and provide relevant interface requirements. (6.2)

430.8

With respect to the design bases for the containment: (6.2)

(1) Discuss the bases for establishing the margin between the maximum calculated accident pressure or pressure difference and the corresponding design pressure or pressure difference. This includes the design external pressure, internal pressure, and pressure between subcompartment walls.

(2) Discuss the capability for energy removal from the containment under various single-failure conditions. State and justify the design basis single failure that affects containment heat removal.

430.9

The Standard Safety Analysis Report (SSAR) states that the analytical models used to evaluate the containment and drywell responses to postulated accidents and transients are included in General Electric Co. report NEDO-20533 and its supplement 1, entitled “The G.E. Mark III Pressure Suppression Containment Analytical Model.” Provide justification that these references are appropriate to use for the ABWR Containment design which is not specified as Mark III. Discuss the similarities and differences of the ABWR design to previously approved Mark II and Mark III designs as they relate to the containment and drywell responses to the postulated accidents and the analytical model used for the analyses. Include in the discussion

the conservatism used in the model and assumptions, the applicable test data that support the analytical models, and the sensitivity of the analyses to key parameters. (6.2)

430.10

With regard to the design features of the containment: (6.2)

(1) Provide general arrangement drawings for the containment structure.

(2) Provide appropriate references to Section 3 of the SSAR which includes the information on the codes, standards, and guides applied in the design of the containment and containment internal structures.

(3) Discuss the possibilities of water entrapment inside containment and its effect on the accident analysis.

(4) Provide information on qualification tests that are intended to demonstrate the functional capability of the containment structures, systems and components. Discuss the status of any developmental tests that may not have been completed.

430.11

Provide a detailed discussion of the likelihood and sensitivity to steam bypass of the suppression pool for a spectrum of accidents. Include in your discussion the following information: (6.2)

(1) A comparison of the ABWR pool bypass capability with that for Mark II and Mark III designs.

(2) The measures for minimizing the potential for steam bypass and the systems provided to mitigate the consequences of pool bypass. Discuss and demonstrate the conservatism of assumptions made in the analysis of steam bypass.

(3) Identify all lines from which leakage (or rupture) could contribute to pool bypass and wetwell air space pressurization.

(4) Identify all fluid lines which traverse the wetwell air space and identify those lines which are protected by guard pipe.

(5) Discuss the rationale and basis for the wetwell spray flow capacity.

430.12

With regard to containment response to external pressure: (6.2)

(1) Describe the wetwell-to-drywell vacuum breaker system and show the extent to which the requirements of subsection NE of section III of the ASME B&PV Code

are satisfied. Discuss the functional capability of the system. Provide the design and performance parameters for the vacuum relief devices.

(2) Discuss the basis for selecting a low design capability for external pressure acting across the drywell to wetwell boundary. It is not apparent that the drywell negative design pressure of 2.0 psid is desirable or sufficient.

(3) The margin between the calculated wetwell-to-reactor building negative differential pressure (–1.8 psid) and the design differential pressure (–2.0 psid) is not considered adequate. A higher margin of 15% should be provided at this stage of the design. Further, given the reliance of the BWR pressure suppression design on containment venting to control pressure, discuss the basis for not providing wetwell-to-reactor building vacuum breakers.

(4) In the analysis of wetwell-to-reactor building negative differential pressure calculation, a 500 gpm wetwell spray flow rate was used. Provide the basis for the assumption and the design basis for the wetwell spray capacity.

430.13

Section 6.2.1.1.3 of the SSAR states that the containment functional evaluation is based upon the consideration of several postulated accident conditions including small break accidents. Provide the assumptions, analysis and results of the small break accidents considered, and demonstrate that the identified (in the SSAR) feedwater line and steam line breaks are the limiting accidents.

430.14

Provide analyses of the suppression pool temperature for transients involving the actuation of safety/relief valves. Provide the assumptions and conservatism employed in the analyses so that an assessment could be made for conformance to the acceptance criteria set forth in NUREG-0783, “Suppression Pool Temperature Limits for BWR Containments.” (6.2)

430.15

Provide the pressure at which the maximum allowable leak rate of 0.5%/day is quoted. (6.2)

430.16

Provide engineered safety systems information for containment response analysis (full capacity operation and capability used in the containment analysis), as indicated in Table 6-7 of Regulatory Guide 1.70, Revision 3. (6.2)

430.17

In the design evaluation section for containment subcompartments (Section 6.2.1.2.3), provide the information necessary to substantiate your assessment that the peak differential pressures do not exceed the design differential pressures. Guidance for the information required is

provided in Regulatory Guide 1.70, Revision 3, Section 6.2.1.2, “Containment Subcompartments”, Design Evaluation.

430.18

Describe the manner in which suppression pool dynamic loads resulting from postulated loss-of-coolant accidents, transients (e.g., relief valve actuation), and seismic events have been integrated into the affected containment structures. Provide plan and section drawings of the containment illustrating all equipment and structural surfaces that could be subjected to pool dynamic loads. For each structure or group of structures, specify the dynamic loads as a function of time, and specify the relative magnitude of the pool dynamic load compared to the design basis load for each structure. Provide justification for each of the dynamic load histories by the use of appropriate experimental data and/or analyses.

Describe the manner by which potential asymmetric loads were considered in the containment design. Characterize the type and magnitude of possible asymmetric loads and the capabilities of the affected structures to withstand such a loading profile. (6.2)

430.19

Provide information to demonstrate that the ABWR design is not vulnerable to a safety relief valve discharge line break within the air space of the wetwell, coupled with a stuck open relief valve after its actuation as a result of the transient. (6.2)

430.20

Discuss suppression pool water makeup under normal and accident conditions. (6.2)

430.21

With respect to mass and energy release analyses for postulated loss-of-coolant accidents identify the sources of generated and stored energy in reactor coolant system that are considered in the analyses of loss-of-coolant accidents. Describe the methods used and assumptions made in calculations of the energy available for release from these sources. Address the conservatism in the calculation of the available energy from each source. Tabulate the stored energy sources and the amounts of stored energy. For the sources of generated energy, provide curves showing the energy release rates and integrated energy release. (6.2)

430.22

In the SSAR sections devoted to containment heat removal systems, identify clearly those areas that may not be part of the GE scope and provide relevant interface requirements. (6.2)

430.23

The SSAR states that the containment heat removal system is designed to limit the long-term temperature of the suppression pool to 207°F. The calculated peak pool temperature is 206.46°F

for the feedwater line break. With respect to this analysis provide the following information: (6.2)

(1) The justification that this is the limiting accident with respect to the maximum temperature in the suppression pool.

(2) The bases for the design margin between the design and calculated temperatures.

(3) All assumptions used in the analysis and conservatism associated with each. Include the effects of potential temperature stratification in the suppression pool and its effects on heat removal capability of the system.

(4) The identification of the decay heat curve used in the analysis.

430.24

Provide the design bases for the spray features of the containment heat removal system. Provide the safety classification of the components associated with the spray feature of the system. (6.2)

430.25

Discuss the rationale for continued reliance on sprays as the sole active engineered safety feature for drywell atmosphere pressure and temperature. Discuss the merits of upgrading the design of drywell fan coolers to provide some capacity for pressure, temperature, and humidity control following an accident. (6.2)

430.26

The time period assumed for initiation of the containment heat removal system after a LOCA is 10 minutes requiring operator action. It is the staff's position that this time period is too restrictive. In fact previous BWR designs (Grand Gulf's Mark III) use 30 minutes actuation time. Provide the reasons why the ABWR does not provide more flexibility with respect to the time required for actuation. (6.2)

430.27

Describe the design features of the suppression pool suction strainers. Specify the mesh size of the screens and the maximum particle size that could be drawn into the piping. Of the systems that receive water through the suppression pool suction strainers under post accident conditions identify the system component that places the limiting requirements on the maximum size of debris that may be allowed to pass through the strainers and specify the limiting particle size that the component can circulate without impairing system performance. Discuss the potential for the strainers to become clogged with debris. Identify and discuss the kinds of debris that might be developed following a loss-of-coolant accident. Discuss the types of insulation used in the containment and describe the behavior of the insulation during and after a LOCA. Include in your discussion information regarding compliance with the acceptance criteria associated with USI A-43 as documented in NUREG-0897. (6.2)

430.28

Provide analyses of the net positive suction head (NPSH) available to the RHR pumps in accordance with the recommendations of Regulatory Guide 1.1. Compare the calculated values of available NPSH to the required NPSH of the pumps. (6.2)

430.29

In SSAR Section 6.2.3, identify clearly those areas that may not be part of the ABWR scope and provide relevant interface requirements. (6.2)

430.30

Provide a tabulation of the design and performance data for the secondary containment structure. Provide the types of information indicated in Table 6-17 of Regulatory Guide 1.70, Revision 3. (6.2)

430.31

Describe the valve isolation features used in support of the secondary containment. Specify the plant protection system signals that isolate the secondary containment and activate the standby gas treatment system. (6.2)

430.32

Identify and tabulate by size, piping which is not provided with isolation features. Provide an analysis to demonstrate the capability of the Standby Gas Treatment System to maintain the design negative pressure following a design basis accident with all non-isolated lines open and the event of the worst single failure of a secondary containment isolation valve to close. (6.2)

430.33

Discuss the design provisions that prevent primary containment leakage from bypassing the secondary containment standby gas treatment system and escaping directly to the environment. Include a tabulation of potential bypass leakage paths, including the types of information indicated in Table 6-18 of Regulatory Guide 1.70, Revision 3. Provide an evaluation of potential bypass leakage paths considering equipment design limitations and test sensitivities. Specify and justify the maximum allowable fraction of primary containment leakage that may bypass the secondary containment structure. The guidelines of BTP 6-3 should be addressed in considering potential bypass leakage paths. (6.2)

430.34

Provide a list of the secondary containment openings and the instrumentation means by which each is assured to be closed during a postulated design basis accident. (6.2)

430.35

Provide a table of design information regarding the containment isolation provisions for fluid system lines and fluid instrument lines penetrating the containment which are within the GE scope of the ABWR design. Include as a minimum the following information:

(1) General design criteria or regulatory guide recommendations that have been met or other defined bases for acceptability;

(2) System name;

(3) Fluid contained;

(4) Line size;

(5) ESF system (yes or no);

(6) Through-line leakage classification

(7) Reference to figure in SSAR showing arrangement of containment isolation barriers;

(8) Location of valve (inside/outside containment);

(9) Type C leakage test (yes or no);

(10) Valve type and operator;

(11) Primary mode of valve actuation;

(12) Secondary mode of valve actuation;

(13) Normal valve position;

(14) Shutdown valve position

(15) Post accident valve position;

(16) Power failure valve position;

(17) Containment isolation signals;

(18) Valve closure time; and

(19) Power source. (6.2)

430.36

For isolation valve design in systems not within the ABWR scope, identify the systems and the relevant interface requirements. Include a discussion on essential and non-essential systems per

Regulatory Guide 1.131 and the means or criteria provided to automatically isolate the nonessential systems by a containment isolation signal. Also, include a discussion on the requirement that the setpoint pressure which initiates containment isolation for nonessential penetrations be reduced to the minimum value compatible with normal operations. (6.2)

430.37
Specify all plant protection signals that initiate closure of the containment isolation valves. (6.2)

430.38
Describe the leakage detection means provided to identify leakage for the outside-containment remote-manual isolation valves on the following influent lines: Feedwater, RHR injection, HPCS, standby liquid control, RWCU connecting to feedwater line, RWCU reactor vessel head spray. (6.2)

430.39
The containment isolation design provisions for the recirculation pump seal water purge line do not meet the explicit requirements of GDC 55 nor does the design satisfy the GDC on some other defined basis as outlined in SRP Section 6.2.4. It is our position that the isolation design in the instance is inadequate and should be modified to satisfy GDC 55 either explicitly or on some other defined basis, with the appropriate justification. (6.2)

430.40
With respect to Figure 6.2-38a

(1) Include the isolation valve arrangement of the standby liquid control system line.

(2) Identify the line labeled in the figure as “WDCS-A” (it joins the RWCU line prior to its connection to the feedwater line), and discuss the isolation provisions for that line.

430.41
Provide a diagram or reference to figure(s) showing the isolation valve arrangement for the lines identified below. For the isolation valve design of each of these lines, provide justification for not meeting the explicit requirements of GDC 56, and demonstrate that the guidelines for acceptable alternate containment isolation provisions contained in SRP 6.2.4 are satisfied. The lines in question are:

- HPCS and RHR test and pump miniflow bypass lines
- RCIC pump miniflow bypass line
- RCIC turbine exhaust and pump miniflow bypass lines
- SPCU suction and discharge lines

430.42

Describe the isolation provisions for the containment purge supply and exhaust lines and discuss design conformance with Branch Technical Position CSB 6-4, “Containment Purge During Normal Operations.”

430.43

Discuss the closure times of isolation valves in system lines that can provide an open path from the primary containment to the environment (e.g., containment purge system). Also discuss provisions of radiation monitors in these lines having the capability of actuating containment isolation. (6.2)

430.44

Identify the system lines whose containment isolation requirements are covered by GDC 57 and discuss conformance of the design to the GDC requirements. (6.2)

430.45

For the combustible gas control systems design, identify clearly those areas that may not be part of the ABWR scope and provide relevant interface requirements. (6.2)

430.46

According to SRP 6.2.5 specific acceptance criteria related to the concentration of hydrogen or oxygen in the containment atmosphere among others are the following:

(1) The analysis of hydrogen ad oxygen production should be based on the parameters listed in Table 1 of Regulatory Guide 1.7 for the purpose of establishing the design basis for combustible control systems.

(2) The fission product decay energy used in the calculation of hydrogen and oxygen production from radiolysis should be equal to or more conservative than decay energy model given in Branch Technical Position ASB9-2 in SRP 9.2.5.

Provide justification that the assumptions used in the ABWR in establishing the design basis for the combustible gas control systems are conservative with respect to the criteria a. and b. above. (6.2)

430.47

Provide an analysis of the production and accumulation of combustible gases within the containment following a postulated loss-of-coolant accident including <u>all</u> applicable information specified in Section 6.2.5.3 of Regulatory Guide 1.70, Revision 3.

430.48

Regarding Containment Type A leakage testing (6.2.6)

(1) Provide the values for P_a and P_t.

(2) Include the acceptance criterion for L_t during preoperational leakage rate tests, i.e., $L_t – L_a$ (L_{tm}/L_{am}), for the case when L_a $(L_{tm}/L_{am}) = 0.7$.

(3) Your acceptance criterion for L_{tm} (SSAR Subsection 6.2.5.1.2.2, Item 1) is at variance with the staff's current practice for acceptance of L_{tm}. Also, it does not comply with the 10 CFR Part 50, Appendix J, Section III, Item A.1.(a) requirement. Therefore, either provide sufficient supporting justification for the exemption from compliance with the above requirement or correct the criterion as appropriate to comply with the requirement. Also, correct the stated acceptance criterion (SSAR Subsection 6.2.6.1.2.2, Item 3) as appropriate to comply with Appendix J, Section III, Item A.6.(b) requirement.

(4) Regarding ILRT, identify the systems that will not be vented or drained and provide reasons for the same.

(5) Provide P&IDs and process flow drawings for systems that will be vented or drained.

430.49

Regarding Type B tests (6.2.6)

(1) Clarify how air locks opened during periods when containment integrity is required by plant’s Technical Specifications will be tested to comply with Appendix J, Section III, Item D.2.(b).(iii).

(2) Provide the frequency for periodic tests of air locks and associated inflatable seals.

(3) Provide the acceptance criteria for air lock testing and the associated inflatable seal testing.

(4) List all containment penetrations subject to Type B tests.

(5) List all those penetrations to be excluded from Type B testing and the rationale for excluding them.

430.50

Regarding Type C tests (6.2.6)

(1) Correct the statement (Subsection 6.2.6.3.1, Paragraph 1) as appropriate to ensure that the hydraulic Type C tests are performed only on those isolation valves that are qualified for such tests per Appendix J. The current statement implies that these tests are not necessarily restricted to the valves that qualify for such tests.

(2) List all the primary containment isolation valves subject to Type C tests and provide the necessary P&IDs.

(3) Provide the list of valves that you propose to test in the reverse direction and justification for such testing for each of these valves.

(4) Identify the valves that you propose to test hydrostatically based on their ability to maintain a 30-day water leg seal. Also, identify other valves which you propose to test hydrostatically and provide the basis for such tests. Provide the test pressure for all the valves mentioned above.

(5) Indicate test pressures for MSIVs (with justification if it is less than P_a) and isolation valves sealed from a sealing system.

(6) Indicate how you will perform Type C leak tests for ECCS systems and RCIC system isolation valves.

(7) Confirm that the interval between two consecutive periodic Type C tests will not exceed 2 years as required by Appendix J.

(8) State what testing procedures you will follow regarding the valves that are not covered by Appendix J requirements.

430.51

Identify the reporting requirements for the tests. Note that your response should address compliance with the requirements in this regard as stated in Appendix J, Sections III.A.(a), IV.A and V. (For example, regarding follow up tests after containment modification, you have not included Type C testing for affected areas). (6.2.6)

430.52

Regarding Secondary Containment (6.2.6)

(1) Identify the special testing procedures you will follow to assure a maximum allowable in leakage of 50 percent of the secondary containment free volume per day at a differential pressure of –0.25" water gauge with respect to the outdoor atmosphere (See Section 6.5.1.3.2).

(2) Identify all potential leak paths which bypass the secondary containment. (For such identification, see (BTP) CSB 6-3, “Determination of bypass Leakage Paths in Dual Containment Plants)”

(3) Identify the total rate of secondary containment bypass leakage to the environment.

430.53

Identify all the interface requirements relating to containment leak testing. (6.2.6)

430.54

Regarding Control Room Habitability systems, (6.4)

(1) Provide the minimum positive pressure at which the control building envelope (which includes the mechanical equipment room) will be maintained with respect to the surrounding air spaces when makeup air is supplied to the design basis rate (295 CFM).

(2) Provide the periodicity for verification of control room pressurization with design flow rate of makeup air.

(3) Clarify whether all the potential leak paths (to be provided in Section 9.4.1) include dampers or valves upstream of recirculation fans.

(4) Identify the action to be taken when there is no flow of the equipment room return fan and consequently the equipment room is over pressurized (Table 6.4-1 contains no information on the above).

(5) Provide the actual minimum distances (lateral and vertical) of the control room ventilation inlets from major potential plant release points that have been used in your control room dose analysis. Also, provide a schematic of the location of control room intake vents.

(6) Provide Figure 6.4-5 (plan view) which you state shows the release points (SGTS vent).

(7) Section 6.4.2.4 and Figure 6.4-1 indicate “only one” air inlet for supplying makeup air to the emergency zone. However, Tables 6.4-2 and 15.6-8 and Section 15.6.5.5.2 indicate that there are “two automatic” air inlets for the emergency zone. Correct the above discrepancy as appropriate. Also describe the characteristics of these inlets with respect to their relative locations and automatic selection control features. State how both flow and isolation in each inlet assuming single active components failure will be ensured.

(8) Describe the design features for protecting against confined area releases (e.g., multiple barriers, air flow patterns in ventilation zones adjacent to the emergency zone).

(9) (i)Describe the specific features for protecting the control room operator from airborne radioactivity outside the control room and direct shine from all radiation sources (e.g., shielding thickness for control room structure boundary, two-door vestibules).

(10) Clarify what you mean by “sustained occupancy” (See SSAR Section 6.4.1.1, Item 3) for 12 persons.

(11) Provide justification for not specifying any unfiltered infiltration of contaminated air into the control room in SSAR Table 15.6-8.

(12) Provide Subsection 6.3.1.1.6 which you state (SSAR Section 6.4.6) contains a complete description of the required instrumentation for ensuring control room habitability at all times.

(13) Give schematics for control room emergency mode of operation during a postulated LOCA (this is required for calculating control room LOCA doses).

(14) The source terms and control room atmospheric dispersion factors (X/Q values) used in the control room dose analysis (See SSAR Tables 15.6-8 and 15.6-12) to demonstrate ABWR control room compliance with GDC 19 are non-conservative. Therefore, reevaluate control room doses during a postulated LOCA using RG 1.3 source terms and assumptions and the methodology given in Reference 4 of SSAR Section 15.6.7. Include possible dose contributions from containment shine, ESF filters and airborne radioactivity outside the control room. Also check and correct as appropriate the recirculation rate in the control room (22.4M ^{3}/sec) given in Table 15.6-8.

(15) Section 6.4.7.1, “External Temperature,” provides design maximum external temperatures of 100°F and –10°F. How are these values used in the design and assessments related to the ABWR? What factors, such as insulation, heat generation from control room personnel and equipment and heat losses, are taken into account? Do these values represent “instantaneous” values or are they temporal and/or spatial averages?

(16) Clarify your position on potential hazardous or toxic gas sources onsite of an ABWR. If applicable, indicate the special features provided in the ABWR design in this regard, to ensure control room habitability.

(17) Identify “all” the interface requirements for control room habitability systems (e.g., instrumentation for protection against toxic gases in general and chlorine in particular; potential toxic gas release points in the environs).

430.55

Regarding ESF Atmosphere Cleanup Systems (6.5.1)

(1) Provide a table listing the compliance status of the standby gas treatment system (SGTS) with each of the regulatory positions specified under C of RG 1.52. Provide justifications for each of those items that do not fully comply with the corresponding requirements. In this context, you may note that the lack of redundancy of the SGTS filter train (the staff considers that filter trains are also active components—See SRP 6.4, Acceptance Criterion II.2.B) is not acceptable. Further, the described sizing of

the charcoal adsorbers based on assumed decontamination factors for various chemical forms of iodine in the suppression pool is not acceptable (RG 1.3 assumes a decontamination factor of 1 for all forms of iodine and RG 1.52 requires compliance with the above guide for the design of the adsorber section). Therefore, revise charcoal weight and charcoal iodine loading given in SSAR Table 6.5-1 as appropriate.

(2) Specify the laboratory test criteria for methyl iodine penetration that will be identified as an interface requirement to be qualified for the adsorber efficiencies for iodine given in SSAR Table 15.6-8. Also, provide the depth of the charcoal beds for the control room emergency system.

(3) Provide a table listing the compliance status of the instrumentation provided for the SGTS for read out, recording and alarm provisions in the control room with “each” of the instrumentation items identified in Table 6.5.1-1 of SRP 6.5.1. For partial or non-compliance items, provide justifications.

(4) Clarify whether primary containment purging during normal plant operation when required to limit the discharge of contaminants to the environment will always be through the SGTS (See SSAR Section 6.5.1.2.3.3). Clarify whether such a release prior to the purge system isolation has been considered in the LOCA dose analysis.

(5) Provide the compliance status tables referred to in Items (a) and (c) above for the control room ESF filter trains. (The staff notes that you have committed to discuss control room habitability system under SSAR Section 9.4.1. However, since evaluation of the control room habitability system cannot be completed until the information identified above is provided, the above information is requested now.)

(6) Identify the applicable interface requirements for the SGTS and the control room ESF atmosphere cleanup system.

430.56

Regarding Fission Product Control Systems and Structures, (6.5.3)

(1) Provide the drawdown time for achieving a negative pressure of 0.25 inch water gauge for the secondary containment with respect to the environs during SGTS operation. Clarify whether the unfiltered release of radioactivity to the environs during this time for a postulated LOCA has been considered in the LOCA dose analysis. (Note that the unfiltered release need not be considered provided the required negative pressure differential is achieved within 60 seconds from the time of the accident.)

(2) Provide justification (See SRP Section 6.5.3, II.4) for the decontamination factor assumed in SSAR Table 6.5-2 and 15.6-8 for iodine in the suppression pool, correct the elemental, particulate and organic iodine fractions given in the tables to be consistent with RG 1.3, and incorporate the correction in the LOCA analysis tables. Alternatively, taking no credit for iodine retention in the suppression pool, revise the LOCA analysis tables. Note that the revision of the LOCA analysis tables (this also includes the control room doses) mentioned above is strictly in relation to the iodine retention factor in the suppression pool (also, there may be need for revision of other parameter(s) given in the tables and these will be identified under the relevant SRP Sections questions).

(3) Identify the applicable interface requirements.

430.57

Regarding SSAR Section 6.7, the staff notes that the Nitrogen Supply System has been discussed under this section, instead of the Main Steam Isolation Valve Leakage Control System (MSIV-LCS) as required by the Standard Format for SARS. The staff will review the material presented in SSAR Section 6.7 along with the material that will be presented in SSAR Section 9.3.1.

Regarding MSIV-LCS, the staff notes that you are committed to provide a non-safety related MSIV leakage processing pathway consistent with those evaluated in NUREG-1169, “Resolution of Generic Issue C-8,” August 1986. Since the staff has not finalized its position so far on the acceptability of the NUREG findings with regard to the design of the MSIV-LCS, provide pertinent information on the system design including interface requirements to evaluate the to-be-proposed design against the acceptance criteria of SRP 6.7. (6.7)

440.75

In the ABWR design, the HPCF is tested by taking suction from and returning water to the suppression pool. Normally the suppression pool water is a lower quality than that of the CST; therefore, draining, flushing and refilling the system is required prior to returning the system to standby after testing. Please discuss the pros and cons of using the CST for testing the HPCF system. (6.3)

440.76

Address the following TMI-2 action items related to ECCS. (6.3)

(1) II.K.1.5

(2) II.K.1.10

(3) II.K.3.17

(4) II.K.3.18

(5) II.K.3.21

(6) II.K3.25

(7) II.K3.30

(8) II.K.3.31

440.77

Confirm that the HPCF system meets the guidelines of Regulatory Guide 1.1 regarding pump Net Positive Suction Head (NPSH). (6.3)

440.78

SRP 6.3 identifies GDCs 35, 36, and 37 in the acceptance criteria. Confirm that the HPCF system, described in Chapter 6.3 of the SSAR, meets the requirements of the above GDCS. (6.3)

440.79

Normally, the HPCF pump takes suction from the Condensate Storage Tank (CST). But, the CST is not seismically qualified or safety related. Confirm that the system piping and level transmitters, which interface with CST, will be designed and installed such that the automatic switchover to the suppression pool takes place without failure. (6.3)

440.80

What is the minimum quantity of water required in the condensate storage tank (CST) for HPCF operation? Give the basis for the required quantity of water in the CST. (6.3)

440.81

What is the closing time of test return valves F009.01C, F011.01C, F015B and F016B? They should close earlier than 36 seconds to prevent any flow diversion to the suppression pool during a LOCA. (6.3)

440.82

Since HPCF is part of the ECCS network, the HPCF pump minimum flow line should be designed to operate for a reasonable length of time. How long can HPCF run in minimum flow mode? (6.3)

440.83

In the resolution of TMI-2 Action Item II.K.3.13, the BWR Owners' Group decided and the staff agreed to keep the initiating RPV level setpoint L2 for starting RCIC and HPCI systems. In ABWR design RCIC is still started at RPV level L2, but he HPCF is started at level 1.5. What is the basis for the initiating level 1.5 for HPCF? (6.3)

440.84

In Section 7.3.1.1.1.2(3) (9) it is stated that HPCF pump discharge pressure is used as a permissive to start ADS automatically. If HPCF is available, ADS may not be required. In the current BWR designs, only low pressure pumps discharge pressures, not HPCS, are used as permissive to start ADS. What is the basis for this change in ADS logic? (63)

440.85

In SSAR Section 7.3.1.1.1.2, it is stated that ADS timer will be set at 29 seconds. Submit the analysis to support the 29 seconds time delay. (6.3)

440.86

SSAR Section 6.3.1.1.4, "ECCS Environmental Design Basis" refers to SSAR Section 3.11 for qualification of ECCS Equipment. However, Section 3.11 does not provide the required information. Provide the necessary equipment qualification information. (6.3)

440.87

Confirm that there are provisions for equipment maintenance during long-term coolant recirculation in the post LOCA environment for ECCS equipment. (6.3)

440.88

Confirm that long-term cooling capacity is adequate in the event of failure of any single active or passive component of the ECCS. (6.3)

440.89

In SSAR Section 6.3.6, References 1 and 4, latest approved revisions of NEDE-29011-P-A are given as references. Identify the latest revisions which are used for ABWR. (6.3)

440.90

In Table 6.3-6 "Plant variables with nominal and sensitivity study values," Item #5 metal water reaction rate, nominal value is given as "EPRI coefficients." Confirm that the EPRI coefficients are the same as used in the model already approved by the staff or identify the EPRI report which discusses these EPRI coefficients. (6.3)

440.91

SSAR Sections 6.3.3.5 and 6.3.3.6 refer to Reference 4 instead of addressing the subjects "use the dual function components for ECCS" and "Limits on ECCS system parameters." Briefly describe the above subjects in the SSAR. (6.3)

440.92

List all computer codes used in the LOCA analysis and give a brief description of each code. (6.3)

440.93

Section 6.3.3.7.2 accident description refers to Reference 4. Provide a brief description of the accident. For details Reference 4 can be used. (6.3)

440.94
Why is there no discharge line fill pump provided for the HPCF system? How does the system design reduce water hammer during the pump start-up? (6.3)

440.95
List the capacity and settings of all relief valves provided for the ECCS to satisfy system overpressure. (6.3)

440.96
Revise SSAR Section 6.3.2.2.1 HPCF to include a description of relief valves provided in the suction and discharge of the HPCF pump. (6.3)

440.97
SSAR Table 5.4-2 gives the design parameters for RCIC system components. Provide similar information for RHR and HPCF systems. (6.3)

440.98
Confirm the 0.099 ft^2 is the lower limit of pipe break size for which ECCS operation is required. (6.3)

440.99
In the RCIC system description (Reference 5.4.6.1.1.1) it is stated that the mixture of the cool RCIC water and the hot steam quenches the steam. Since RCIC is injected to the reactor through the feedwater system, this statement may not be true. (6.3)

440.100
In the remote shutdown system RCIC controls are replaced by HPCF controls. Traditionally, RCIC was used for remote shutdown because the system will be available during station blackout. Describe the basis for replacing RCIC controls with RPCF controls in the remote shutdown panel. (6.3)

20.2.7 Chapter 7 Questions

420.1

Identify the topical reports that will be provided to support any aspects of the design that are substantially different relative to designs previously reviewed by the staff. Subjects addressed in these topical reports should include but not necessarily be limited to the following:

Overall block diagram(s) and descriptions of the reactor protection and engineered safety features actuation system, showing the architecture of the system, the allocation of functions to modules, and the communication channels among modules. Digital and analog modules should be identified. Methods for assuring required independence should be clearly identified, as well as power supply dependencies, division boundaries and non-safety system interfaces. A description of the scope of on-line and diagnostic testing features for the proposed system should be provided with reference to this diagram, to illustrate compliance with testability requirements. (7)

420.2

Identify the topical reports that will be provided to support any aspects of the design that are substantially different relative to designs previously reviewed by the staff. Subjects addressed in these topical reports should include but not necessarily be limited to the following:

The applicant's overall design verification program, covering development of the functional requirements, criteria, specifications, design, manufacture, test, and qualification methods and procedures; this should include a V&V plan for software design verification/validation. (7)

420.3

Identify the topical reports that will be provided to support any aspects of the design that are substantially different relative to designs previously reviewed by the staff. Subjects addressed in these topical reports should include but not necessarily be limited to the following:

Failure modes and effects analysis for the I&C system.

420.4

Identify the topical reports that will be provided to support any aspects of the design that are substantially different relative to designs previously reviewed by the staff. Subjects addressed in these topical reports should include but not necessarily be limited to the following:

A defense-in-depth analysis, demonstrating the diversity in the system that precludes the likelihood of common mode failures.

420.5

Identify the topical reports that will be provided to support any aspects of the design that are substantially different relative to designs previously reviewed by the staff. Subjects addressed in these topical reports should include but not necessarily be limited to the following:

System (and significant component) reliability goals, assumptions, methodology, model, analysis, and evaluation.

420.10

Identify the topical reports that will be provided to support any aspects of the design that are substantially different relative to designs previously reviewed by the staff. Subjects addressed in these topical reports should include but not necessarily be limited to the following:

Task analysis for the man/machine interface to the system.(7)

420.11

Identify the topical reports that will be provided to support any aspects of the design that are substantially different relative to designs previously reviewed by the staff. Subjects addressed in these topical reports should include but not necessarily be limited to the following:

Wide Range Neutron Monitor design basis. (NEDO-31439, May 1987) If this system is not part of the ABWR (Section 7.6.1.1 indicates it is not) provide justification for its exclusion.(7.6.1.1)

420.12

Identify the topical reports that will be provided to support any aspects of the design that are substantially different relative to designs previously reviewed by the staff. Subjects addressed in these topical reports should include but not necessarily be limited to the following:

10CFR50.62 (ATWS) conformance. Specifically address the manually initiated SLCS conformance (7.4.2.2.2(1)) to the ATWS rule (50.62(4)) of automatic initiation. (7.4.2.2.2)

420.13

One of the goals of the ABWR is simplification. The October, 1987 presentation mentions a 60% reduction in instrumentation. Which plants is this referenced to? Provide a description of the instrumentation which is no longer considered necessary.(7)

420.14

Address the effects of Station Blackout on the HVAC required to maintain functional electronics.(7.1.2.3.9)

420.15

Address the redundancy and diversity of the power supplies for ARI. (7.4)

420.16

Address the decision to make the ARI non-1E instead of 1E system. (7.4)

420.17

Describe the trade-off analyses leading to the selection of an analog or digital approach for implementing the logic of the safety system. Describe the major criteria that the tradeoff was based on. Show how the tradeoff criteria is in accordance with applicable design criteria.(7)

420.18

For the proposed use of digital computers, show how the digital system is superior to analog alternatives to implementing the logic. Show how the analyses determined that the reliability of the digital computer based system was better than the reliability of the analog system. (7)

420.19

This section states that automatic self-test is performed sequentially on all four divisions, to minimize common mode effects, and that a complete self-test sequence through all four divisions takes no more than 30 minutes. The original response to Question 19 revised this section. What hardware and software design features are provided to allow sequencing and testing of the four divisions without violating independence/isolation criteria? The revised section appears to allow a common centralized test driver. Illustrate with a block diagram. (7.1.2.1.6.(4))

420.20

Describe the fiber optic links in the safety systems. What signals are multiplexed on each link? Show how the independence criteria in accordance with IEEE-603 and IEEE-379 is satisfied with the proposed configuration of fiber optic links.(7)

420.21

Describe the safety computer system's interface to any non-safety computer systems and other plant instrumentation. Describe if information transfer from 1E to N-1E computers is via broadcast or handshake. (7)

420.22

Provide a table of conformance to IEEE-603 and ANSI/IEEE-7-4.3.2. (7)

420.23

Provide a table of conformance to IEEE-384, indicating where credit is taken for isolation or separation, what devices or methods are used, and the basis of isolation device qualification. If specific types of components have not been chosen, provide specification level information including testing acceptance criteria. (7)

420.24

Are any artificial intelligence features provided in the proposed system, whereby probabilistic judgements are made by the system, or whereby the system can "learn" during its operational life? (7)

420.25

Is credit taken in the safety analysis for any rotating memory devices such as disk drives? (7)

420.26

What is the definition of "Safety Associated" as used in SAR Section 7.1.2.1.6? (7.1.2.1.6)

420.27
Specify which parameters are to be triplicated. At what point does the triplication start (flow orifice, sensor?) and end (transmitter, trip logic?). If there is triplication of sensors is there diversity between sensors? (7)

420.29
For those systems where it has not already been done (example 7.1.1.3.5) clarify whether manual or automatic initiation will be used. (7.1.1)

420.30
Define the word “sufficient” used in section (j). (7.1.2.2)

420.31
For section 7.1.2.3.2(1)(c,d,e) and (2)(a) define “sufficient”. (7.1.2.3.2)

420.32
The listed design basis should include instrumentation necessary to inform the operator that isolation has been completed and control should provide ability for operator to reset (with adequate safeguards against inadvertently breaking isolation). (7.1.2.3.2)

420.33
Add to 7.1.2.3.2(2)(c)... “without causing plant shutdowns” or reducing safety margins. (7.1.2.3.2)

420.34
For Section 7.1.2.3.7(1)(b) provide a listing of the nonessential parts of the cooling water system which should be isolated. List any nonessential parts for which isolation is not provided. (7.1.2.3.7)

420.35
Is the wetwell to drywell vacuum breaker control manual or automatic? (7.1.2.6.5)

420.36
If the CAMS system is only a monitoring system, why is it not always on instead of waiting for a LOCA to monitor radiation? (7.1.2.6.6)

420.37
What is the immediate safety action required by relief valve leakage and is it automatic? (7.1.2.6.7)

420.38
The table indicates RG 1.151 applies only to safety related display and Non-1E control systems. Section 7.1.2.10.11 refers to other safety systems including RPS and ECCS. Clarify which systems RG 1.151 is to apply to. (Table 7.1-2)

420.39

The table lists few systems for which RG 1.97 is applicable. Address the RG 1.97 for all categories and variables. (Table 7.1-2)

420.40

The HPCF pump is interlocked (7.3.1.1.1.1(3)(c)) with the undervoltage monitor. If the breaker cannot close will it retry and what information is available to the operator if it doesn't close that would indicate an undervoltage problem? (7.3.1.1.1.1)

420.41

Does the 36 seconds (7.3.1.1.1.1(3)(e)) include time for diesel generator to start? (7.3.1.1.1.1)

420.42

Section 7.3.1.1.1.1(3)(f) states that separation prevents a single design basis event from disabling core cooling. This section should note that this event must be considered in conjunction with an additional single failure. (7.3.1.1.1.1)

420.43

Manual pushbuttons are provided to initiate ADS immediately if required. Describe when manual action is required before the 29 second timer actuates ADS. (7.3.1.1.1.2(3)(c))

420.44

One pressure sensor is used to detect low RCIC system pump suction pressure. Explain the criteria used to justify a single pressure sensor. (7.3.1.1.1.3(4)(a))

420.45

Define analog indication. Is this an analog system or digital simulation? (7.3.1.1.1.3(6))

420.46

The injection valves cannot be opened at normal pressure. Is this because of interlocks or because of motor size? (7.3.1.1.1.4(3)(g))

420.47

Is the suppression pool cooling automatically initiated? The SAR describes the system as being used to reduce the suppression pool temperature immediately after a blowdown. Section 5.4.7.1.1.5 indicates automatic initiation. (7.3.1.1.4)

420.48

SAR 7.1.2.1.6(2) appears to define “fault” as the “...inability to open or close any control circuit.” Explain the basis for this definition and the extent of its use in the FMEAs. Are there any other potential failure modes excessive time to close a circuit? (7.1.2.1.6)

420.49

Describe the fault tolerant features of the digital design. Describe the types of faults that are tolerated by these design features. Show how these features would respond to various faults, and show that the effectiveness of the safety system is not compromised. (7)

420.50

Describe the self-diagnostic features of the computer-based safety system. Describe the diagnostics that are run on-line, in a background mode and in a maintenance mode. Describe what happens when an on-line diagnostic uncovers an error in the computer system. (7.1)

420.51

Describe the data buses that are used in the multiplexers. Describe the features that are implemented to ensure that the bus or multiplexer is not cause of a single point failure. Describe what happens when a single card on a data bus fails. Show what design features prevent the error from propagating and not challenging the remainder of the safety system. If specific equipment has not been selected, please provide the interface criteria. (7.1)

420.52

As indicated in the October 1987 ABWR presentation, the self-test sequence of the digital processor equipment is supposed to reduce the need for surveillance and monitoring by human personnel. Describe how it was proven that the old and new surveillance schedules are functionally equivalent. (7)

420.53

Is a diverse (hardware implemented) watchdog timer provided in the design for detecting system stall? (7)

420.54

Does the FMEA consider unusual failure modes and their effects such as system stall, interruption and restoration of power (or function), metastability, or timing errors? Provide a descriptive summary of the failure modes addressed in the FMEA or describe the interface criteria. (7)

420.55

Provide a summary of any graceful degradation features provided in the I&C systems or describe the interface criteria. (7)

420.56

Demonstrate that the effects of hardware and external failures on software performance have been sufficiently addressed in the FMEA or describe the interface criteria. (7)

420.57

What provisions have been made in the design process to preclude the introduction of a software virus that could affect the system when operational? (7)

420.58

Beyond the redundancy requirements levied by single failure criteria, provide information to demonstrate sufficient diversity in the I&C system to preclude common mode failures. (7)

420.59

Describe the methods which are used to assure that equipment which is not qualified for all service conditions will not spuriously operate during exposure to conditions for which the equipment is not required to function to mitigate the effects of accidents or other events. (7)

420.60

Provide examples for section (g) which meet the design bases. (7.12-2)

420.61

Explain section (h) further. Does this mean one 480V bus, 4160 bus the generator? Same question at 7.2.3.2(2)(b). (7.1.2.2)

420.62

Provide justification for going to a 2/3 scram instead of 1/3 when one is bypassed. (7.1.2.10-11)

420.63

What are the reliability/availability goals for the reactor protection and engineered safety features systems? (7)

420.64

Describe the reliability model and assumptions used to demonstrate achievement of the reliability goals; this should include a description of the system architecture. (7)

420.65

What methodology is used in determining the system reliability/availability? (7)

420.66

Describe the data validation features in triplicated sensors. (7)

420.67

What testing will be done to demonstrate reliability? What is the specific scope of these tests? (7)

420.68

What is the effect upon the number of spurious trips generated by the RPS if the digital design replaces the previous analog design? Provide comparison. (7)

420.69

Are there any limitations on the ABWR design concerning the use of expert systems? Any limitations on the use of technology not specifically described? The original response does not describe an approach for determining what hardware or software developments which may

occur between design certification and plant operation can be implemented without changes to the design certification and NRC review. (7A)

420.70
Is there any system for in-service testing of the ARI? (7.1.2.1.6)

420.71
Is the CRD scram discharge high water level used as the example of the fifth test valid given that there is no scram discharge volume? (7.1.2.1.6)

420.72
Section (1) of 7.1.2.1.6 states that normal surveillance can identify failures. Discuss whether this system has the capability of transmitting this information to the plant computer so that an immediate alarm can be given in addition to waiting for the scheduled surveillance. (7.1.2.1.6)

420.73
Section (4) notes that the four divisions are tested in sequence. When the thirty minute sequence is complete does the test system start over again or is this an operator initiated test? (7.1.2.1.6)

420.74
Section (5) notes that only one division shall be bypassed at any one time. Describe the interlock protection or administrative controls which assure this. (7.1.2.1.6)

420.75
For section 7.1.2.2(j) clarify that the physical and electrical separation does not preclude the proper environmental qualification of redundant I&C equipment. (7.1.2.2)

420.76
For section 7.1.2.3.2(1)(c,d,e) and (2)(a) define “sufficient”. (7.1.2.3.2)

420.77
One of the reasons stated for the utilization of microprocessors for the implementation of instrumentation and logic functions is that less uncertainty exists in the margins between actual safety limits and the limiting safety trips. The margins are stated to be set from experimental data on setpoint drift (see Section 7.1.2.1.4.1) and from quantitative reliability requirements for each system and its components.

Provide the documented bases for this procedure. (7.1.2.1.4.1)

420.78
One of the reasons stated for the utilization of microprocessors for the implementation of instrumentation and logic functions is that less uncertainty exists in the margins between actual safety limits and the limiting safety trips. The margins are stated to be set from experimental data on setpoint drift (see Section 7.1.2.1.4.1) and from quantitative reliability requirements for each system and its components.

Will this procedure be a topical report used as a design tool? (7.1.2.1.4.1)

420.79

One of the reasons stated for the utilization of microprocessors for the implementation of instrumentation and logic functions is that less uncertainty exists in the margins between actual safety limits and the limiting safety trips. The margins are stated to be set from experimental data on setpoint drift (see Section 7.1.2.1.4.1) and from quantitative reliability requirements for each system and its components.

What experimental data has been used to provide inputs to this design approach? (7.1.2.1.4.1)

420.80

Section 7.1.2.3.1(1)(c) states that no operator action is required for 10 minutes following LOCA. Section 6.3.1.1.1(3) states, that no operator action is required for 30 minutes after an accident. Section 6.3.2.8 also states 30 minutes. Clarify which statement is the design basis. Same question @ 7.3.1.1.1.4(3)(i) and 7.3.1.1.1.2(3)(i). (7.1.2.3.1)

420.81

Section 7.1.2.3.1(1)(c) states that operator action is not required. Describe what operator actions are desired but not required for the first period of time (10 or 30 minutes) for various accident scenarios. (7.1.2.3.1)

420.82

In section 7.1.2.3.3(1)(c) is manual control required only after 30 minutes? Why isn't automatic control also provided? (7.1.2.3.3)

420.83

Is the suppression pool cooling also provided with automatic control? (7.1.2.3.4)

420.85

How are the Class 1E circuits protected/isolated from the 1E and N-1E CRT high voltage circuits in the main control panels? (7)

420.86

If hardwired meters are used explain how the adjacent electronics in the control panels are protected from EMI and fault propagation from faulted current transformers. (7)

420.87

The response noted that RIP trips have mostly been caused by noise in the adjustable speed drive (ASD). Describe the changes that have been made to reduce the susceptibility of the RIP's or the reduction in noise of the ASDs. (7)

420.88

List the criteria or standards for surge withstand capability to be applied to the equipment. ANSI/IEEE-C62.45-1987 “Guide on Surge Testing for Equipment Connected to Low-Voltage

AC Power Circuits” is an example of criteria currently being applied to limit the possible affects from line surges. (7)

420.89

List the design goals for the survivability and continued operation of safety systems equipment in the presence of line switching transients, lightning induced surges and other induced transients within the systems as installed. (7)

420.90

Address the possible effects of electrostatic discharge (ESD) at keyboards, keyed switches and other exposed equipment components. (7)

420.91

Most of the I&C system microprocessor equipment is likely to be located in a mild environment, but survivability requirements or limitations on the voltage potential buildup by humidity control or other measures is not discussed. Also, the data concentrators are provided at remote locations where the environmental control is not clearly described. Identify the criteria, design limits and testing program for this area of ESD controls. (7)

420.92

The application of high technology semiconductor materials and related technologies to computing devices has resulted in high current densities in some portions of equipment used in non-nuclear applications. This type of equipment may be used for the ABWR.

Identify how these higher current densities, which can result in localized high heat spots, will be considered in the design described by Section 7.0. (7)

420.93

The application of high technology semiconductor materials and related technologies to computing devices has resulted in high current densities in some portions of equipment used in non-nuclear applications. This type of equipment may be used for the ABWR.

Does an analysis of these potential hot spots result in special thermal design constraints? (7)

420.94

The application of high technology semiconductor materials and related technologies to computing devices has resulted in high current densities in some portions of equipment used in non-nuclear applications. This type of equipment may be used for the ABWR.

What design criteria are to be applied and what will be the effects upon the microprocessor reliability? (7)

420.95

The application of high technology semiconductor materials and related technologies to computing devices has resulted in high current densities in some portions of equipment used in non-nuclear applications. This type of equipment may be used for the ABWR.

Since the plant environmental limitations only identify general area temperature ranges, what consideration will be given to localized cooling and heat transfer? (7)

420.97

This refers to Section 3.11 for EQ. Section 3.11 invokes IEEE-323 as a basis for qualification. IEEE-323 was written assuming 40 year life. Address how this standard is to be extrapolated to a 60 year design life for the ABWR. (7.3.1.1.4(h))

420.98

The more extensive use of semiconductors and fiber-optic materials in the RPS identifies an area of design not previously discussed in the standard review plan. The radiation qualification for semiconductor and fiber-optics is an evolving part of the technology related to microprocessors and fiber-optic communication networks.

Since the semiconductor and fiber-optic materials are to be distributed throughout the plant, the staff requires that the criteria and design application details be identified in order that equipment reliability and operating life projections may be identified. Space, defense and airline applications have developed criteria and standards which may apply to the ABWR. The October 1987 presentation identified the airline industry as a source of established technology for intelligent multiplexing systems.

Identify the specific airline criteria and standards which will form a part of the design guidance and list any other sources that GE is using as guidance. (7)

420.99

While a computer-based system can provide more effective man/machine interface, the internal system operation is more complex, and can be more obscure to the operator or maintenance person if he is required to intervene at a complex level.

Have the operator tasks with regard to interfacing with the safety system been analyzed? What was the result of the analysis? How did the result of the analysis affect the requirements, design and implementation of the safety system? (7)

420.100

While a computer-based system can provide more effective man/machine interface, the internal system operation is more complex, and can be more obscure to the operator or maintenance person if he is required to intervene at a complex level.

Describe the hardware design features that provide administrative control of devices capable of changing the data or program in the computer-based safety system. (7)

420.101

While a computer-based system can provide more effective man/machine interface, the internal system operation is more complex, and can be more obscure to the operator or maintenance person if he is required to intervene at a complex level.

What data or program elements are adjustable/selectable by the operator? (7)

420.102

While a computer-based system can provide more effective man/machine interface, the internal system operation is more complex, and can be more obscure to the operator or maintenance person if he is required to intervene at a complex level.

What capability of providing a permanent and current record of the system data base is provided in the system? (7)

420.103

While a computer-based system can provide more effective man/machine interface, the internal system operation is more complex, and can be more obscure to the operator or maintenance person if he is required to intervene at a complex level.

Provide the basis for assumed operator response times. (7)

420.104

While a computer-based system can provide more effective man/machine interface, the internal system operation is more complex, and can be more obscure to the operator or maintenance person if he is required to intervene at a complex level.

Discuss the range of possible scenarios for transferring the system from automatic to manual mode (and vice versa) and the potential for error or disturbance during such a transfer.

Describe any differences characterized by these transfers with respect to BWR designs previously reviewed by the staff. For example, discuss consideration of I&E Bulletin 80-06, “Engineered Safety Features Reset Controls”. (7)

420.105

The current criteria for ATWS capabilities is the NRC ATWS Rule 10CFR50.62. The existing BWR plant designs have been provided with a Safety Evaluation of the Topical Report (NEDE-31096-P) which contains an Appendix A “Checklist for Plant Specific Review of Alternate Rod Injection System (ARI).” No topical reference was found in the submittal.

Indicate if this checklist is applicable to this design and how the compliance to the ATWS rule is to be achieved. (7)

420.106
Define the logic by type and verify the diversity of the reactor internal pump trip circuits. If software is to be a part of this design, identify the form and diversity to be applied to this function. (7)

420.108
In section (m) consider replacing “obviate” with prevent or preclude. (7.1.2.2)

420.109
In Section 7.1.2.3.1(c), describe how provision for manual control limits dependence on operator judgement in times of stress. (7.1.2.3.1)

420.110
For Section 7.1.2.3.1(2), describe any precautions taken to prevent or minimize inadvertent initiation of non-safety systems during accidents. (7.1.2.3.1)

420.111
Why isn't the requirement to meet the Seismic Category I design requirements (7.1.2.3.7(1)(c)) listed in the other applicable sections? (7.1.2.3.7)

420.112
Are the other sections to be revised to include the normal operation parameters similar to 7.1.2.4.3(1)(a)? (7.1.2.4.3)

420.113
Has consideration been given to providing the annunciators with backup diesel or battery power? (Ref. 7.1.2.6.1.1(2)(g)). (7.1.2.6.1.1)

420.114
The copy of Section 7 provided to the staff did not include Appendix 7A nor an indication that it was to be provided later. Provide this section or a schedule for providing it. (7A.1-1)

420.115
In the discussion about torque switches and thermal overloads, there is a reference to Section 3.8.4.2 which is the applicable codes and standards for seismic qualification of the Reactor and Control Buildings. What is the correct reference? (7.3.1.1.1.3(4)(e))

420.119
Are there any other valves which must isolate upon initiation of the SLCS? (7.4.1.2(7))

420.120
List all exemptions to the requirement rather than providing an example. (7.3.2.1.2(3)(c))

420.121

The first paragraph states that pipe break outside containment and feedwater line break are discussed below. The staff could not locate these items. (7.3.1.2(7))

420.122

Is the instrumentation required for the operator to verify bypass valve performance and relief valve operator 1E or N-lE? (15.2.2.2.1.4)

420.123

SSAR 15B.4 describes the essential multiplexing system (EMS) in some detail. SSAR Figure 7A.2-1 states that the design is not limited to this configuration. It is our understanding that the EMS design is still in a preliminary design stage. Is SSAR 15B.4 still accurate and is the design limited to that configuration? (15B.4)

420.124

The FMEA submitted in SSAR 15B.4 is inadequate for a safety evaluation supporting the design certification. The FMEA appears to the staff to be oversimplified with one line item each for component failures and does not address potential software complications. The staff requests clarification of how this FMEA was developed given that the system design has not been finalized. The staff also believes that software failures need to be evaluated. The failure modes investigated should include, as a minimum, stall, runaway, lockup, interruption/restoration, clock and timing faults, counter overflow, missing/corrupt date, and effects of hardware faults on software. (15B.4)

420.125

This section provided additional clarification of the intended use of the remote shutdown system. The degree of independence and isolation from the Safety System Logic and Control (SSLC) and EMS are not clear. Is it intended in the SSAR to take credit for the RSS if there is a total loss of EMS? (7.4.1.4)

420.126

Compared with GESSAR II, the ABWR has significantly reduced the number of input sensors by use of sharing sensors. Provide a bases to why this does not increase potential vulnerability to common mode failures by reducing sensor diversity. (7A-7)

420.127

In general, the applicant should provide a clear presentation of how the ABWR with common software and hardware modules for many functions (including SSLC logic self-test programs) conforms with IEEE-279-1971 and is at least as single failure proof as GESSAR II. The discussion of shared sensors in 7A-7 does not address potential common mode software failures which may be capable of defeating the diverse parameters. Additionally, the applicant should address why diversity of software should not be a requirement to maintain system diversity. (7)

420.128
Will software be used to isolate data? If so, what are the design and qualification criteria that are to be applied? Are there any systems which have non-Class IE software such as keyboard or display control software that interface with the Class-lE systems? Are there any interface with the Class-lE systems which receive inputs from non-Class-lE systems or other channels of 1E systems. (7A.7)

420.129
List those systems or major components in the I&C design area for which the design is not complete to the “purchase specification” level. (7)

420.130
In response to Question 420.63, a MTBF goal of 100,000 hours (11.4 years) is given for the essential multiplexing system. Is this goal for one channel or the complete system? If this goal is for the complete system, it appears to the staff that the ABWR can expect to loose control at the control room of many of the safety systems (RPS, RHR, ADS) five or six times over the lifetime of the plant. How does this compare with the reliability/availability of multiple ESF systems in the BWR/5 & 6 design (or GESSAR II)?

420.131
Are multiplexer and software failures included in these systems interactions and common cause failures? (19.2.3.4)

420.132
Section 19.3.1.3.1 (b) states that “if core cooling is accomplished without the use of an RHR systems and the suppression pool cooling begins overheating, the suppression pool cooling mode of the RHR will be initiated by the operator.” Is any manual action required prior to 30 minutes? (19.3.1.3.1 (b))(Response 420.47)

420.133
Subsection 19.3.1.3.1(c)(i) describes the MSIV closure sequence with the most desirable outcome requiring operator action at 30 seconds to insert rods. If that fails the operator must inhibit ADS valves from opening and initiate SLCS within 10 minutes. These activities do not appear to be consistent with stated design goal of no operator action for 30 minutes following a transient. Provide a description of how the MSIV closure sequence meets the 30 minute rule (6.3.1.1.1) same question for Loss of Offsite Power (LOOP).

420.134
Equipment maintenance or test unavailability are taken from GESSAR PRA and are based upon BWR experience. In the past, I&C has been a large contributor to system downtime. How do these systems (RHR, RCIC) unavailability numbers take into account the new multiplexing and microprocessors? (19D.3.4)

420.135

Provide the justification for Mean Time To Repair (MTTR) of 4 hours for multiplexers and 30 minutes for ESF logic. Inverters and battery chargers have restoration time given in (Table 19A.8) as 48-56 hours. Are the multiplexers designed with all test and maintenance equipment installed? (Table 19D.6-10)

420.136

The staff has reviewed the commitments in the SSAR and has reviewed the available documentation describing the verification and validation plans. To date, the information has been vague, general in nature and lacking in essential detail to demonstrate conformance with ANSI/IEEE-7-4.3.2. Does the applicant intend to enclose the V&V Plan as Appendix B of SSAR Chapter 7 or will the V&V details be left as an interface requirement? The staff requires a formal, structured V&V plan to be in place and implemented early in the software design process. (7A)

20.2.8 Chapter 8 Questions

435.1

The scope of electrical systems that GE intends to provide under the ABWR design is poorly defined. In Subsections 1.2.2.5.1.1 and 8.1.2.1 a brief description of the Unit Auxiliary AC Power System is provided that states that this system supplies power to unit loads that are non-safety related and uses the main generator as the normal power source with the reserve auxiliary transformers as a backup source. It is not clear however whether this system will be provided under the ABWR design. No detailed description or single line diagrams of this system, the main generator, unit auxiliary transformer, or reserve auxiliary transformers are provided. The staff requires that a clear distinction be made between the electrical systems that will be provided under the scope of the ABWR standard design and those that will be provided by others. This is necessary so that the staff can judge the completeness and adequacy of the electrical systems within the ABWR design and the completeness and adequacy of the interface requirements to those systems outside the ABWR design scope. Please provide this information.

435.2

The ABWR SSAR does not address how the ABWR will cope with a station blackout event. The station blackout rule, 10CFR 50.63, which became effective July 21, 1988, requires that each light-water-cooled nuclear power plant licensed to operate must be able to withstand for a specified duration and recover from a station blackout (loss of all alternating current power). Please provide details on the design aspects of ABWR systems and equipment that will be used to cope with a station blackout. In particular address the capabilities of the DC power systems to cope with a station blackout, the loading and endurance of the batteries used to cope with a station blackout, and the capabilities of any alternate AC (AAC) power sources used to cope with a station blackout. Identify any interface requirements needed on the offsite power system or other systems in order to support the station blackout design criteria. Additional information and guidance on station blackout can be found in Regulatory Guide 1.155 and NUMARC-8700.

435.3

Section 8.1.2.1 of the ABWR SSAR states that the transfer of the Class 1E buses to the alternate preferred power source is a manual transfer. This seems to contradict Subsections 3.1.2.2.9.2.1 and 3.1.2.2.9.2.2 which indicate that transfer is automatic. Please clarify, and if the transfer is automatic provide the details on the type of transfer (slow, fast, make-before-break, etc.), the signals used to initiate transfer, how the transfer is accomplished.

435.4

(1) In section 8.2.3 of the ABWR SSAR one of the Nuclear Island interfaces identified is four 6.9 kV feeders to four transformers powering ten RIP pumps. However, figure 8.3-1 and figure 8.3-2 show motor generator sets between two of the 6.9kV feeders and the RIP pumps. Please clarify whether the motor generator sets will be used in the ABWR design and if so. Describe their function.

(2) Also, with regard to the same subject, section 15.3.1.1.1 states that since four buses are used to supply power to the RIPs, the worst single failure can only cause three RIPs to trip, and the frequency of occurrence of this event is estimated to be less than 0.001 per year. Further down in this same section a statement is made that the probability of additional RIP trips is low (less than 10^{-6} per year). Justify these figures in light of the fact that historically, a total loss of offsite power occurs about once per 10 site-years (NUREG/CR-3992). Also, has the effect of a fault on the common feeder upstream of the 6.9kV feeders been considered with respect to the coastdown capability of the RIPs and motor generator sets (braking effect).

435.5

(1) Section 8.2.3 identifies the normal voltage and number of feeders interfacing between the Nuclear Island and remainder of plant [power systems; but they do not specify any interface requirements such as voltage and frequency tolerances, available fault current, loading. availability, etc. that are necessary to completely define the required interfaces. Please provide the information.

(2) You also need to provide additional information on the power sources *Unit Transformer, Startup Transformer, etc.) and the way they are configured to provide power to the RIP pumps in order to support the availabilities claimed for these power sources in section 15.3.1. We suggest a one-line diagram similar to that which you provided in your presentation to the staff on September 14, 1988, be included in the ABWR SSAR to better define this interface.

435.6

Subsection 8.3.1.1.4.1 and Figure 8.3-4 briefly describe the 120 VAC Safety-Related Instrument Power System. This is interruptible power backed up by the divisional diesel generators. Please identify the major loads and type of instrument loads fed by this system.

435.7

Subsection 8.3.1.1.4.2.2 and Figure 8.3-6 briefly describe the Class 1E RPS Power Supply. They show a rectifier and inverter fed from the 480 VAC Class 1E power system which is backed up by the 125 VDC power system. They do not however show an independent electrical protection assembly (EPA) on the output of the RPS power supply. Redundant EPAs were required (September 24, 1980 letter to all operating BWRs) on the output of past non-Class 1E RPS power supplies in order to satisfy the single failure criteria for non-fail-safe type failures (undervoltage, overvoltage, underfrequency). Because a Class 1E RPS power supply is used on the ABWR, redundant EPAs are not required since failure of the Class 1E supply is the first random failure taken. However, because that failure could be a non-fail-safe type failure that

could result in loss of the scram function, at least one independent EPA should be monitoring the output of the RPS power supply.

(1) Please describe the type of EPA that will be used and discuss its independence from the RPS power supply.

(2) Also provide the voltage and frequency setpoints and tolerances that will be used on the EPA.

435.8

Section 8.3 does not identify any interfaces between the Nuclear Island and the remainder of plant systems within the onsite power systems. Please verify that all of the onsite power systems are within the Nuclear Island scope, or identify the interfaces and the interface requirements.

435.9

Section 8.3.1.1.4.2.3 and Figure 8.3-5 briefly describe the Process Computer Constant Voltage, Constant Frequency Power Supply; but they do not state whether it is qualified Class 1E, although it is discussed under Section 8.3.1.1.4.2 entitled “120V AC Safety Related Uninterruptible Power Supplies (UPS).” The backup to this power supply is from the non-Class 1E 250 VDC battery, and Section 8.3.2.1 states that all of the 250 VDC loads are non-Class 1E.

(1) Please clarify whether the Process Computer Power Supply is qualified Class 1E.

(2) If it is Class 1E explain why a backup non-Class 1E 250 VDC supply is connected to it, and describe the Class 1E/non-Class 1E isolation provided.

(3) If it is non-Class IE explain why a normal and backup Class 1E 48 VAC supply is connected to it, and describe the Class 1E/non-Class 1E isolation provided.

435.10

(1) Section 8.3.1.1.4.2.4 states that the function of the Vital AC Power Supply System is to provide reliable 120V uninterruptible AC power for important non-safety related loads that are required for continuity of power plant operation. However it does not identify the non-safety related loads that it supplies, nor is a one-line diagram of the power supply system provided. Please identify the non-safety related loads that this system supplies and include a one-line diagram of the power supply system in the ABWR SSAR identifying the power sources to it. If there are any 1E/non-lE interfaces identify the isolation provided.

(2) This section also states that an independent 125V DC system, including a battery and battery charger, is the normal Source of power for the Vital AC Power System. However section 8.1-2.1 states that there are no non-Class 1E 125 VDC batteries

supplied as part of the plant design. Please clarify this apparent discrepancy. Also, include this system in the one-line diagram to be provided for the Vital AC Power System.

435.11

Section 8.3.1.1.5.1 describes the physical separation and independence of electrical equipment and wiring. It seems to indicate that there is separation between the divisions but a statement is made that seems to imply that the separation may not at in all cases be total. This statement says that electric equipment and wiring for the Class IE systems which are segregated into separate divisions are separated so that no design basis event is capable of disabling any ESF total function. This statement could be interpreted to mean that in an area with three divisions, each with 100% capability, a single design basis event would be allowed to fail two of the divisions since 100% capability for the ESF function would still survive. Please clarify this point and indicate whether a single design basis event will ever be allowed to fail more than one division.

435.12

Design criteria (4) in section 8.3.1.1.5.2 states that interrupting capacity of switchgear, load centers, motor control centers, and distribution panels is compatible with the short circuit current available at the Class 1E buses. Verify that this criteria ensures that the interrupting capacity of this equipment will be equal to or greater than the maximum available fault current to which it could be exposed.

435.13

The first statement in section 8.3.1.1.6.4 indicates that the only protective trips active on the diesel generators during LOPP or LOCA conditions are the generator differential relays and the engine overspeed trip device. Following statements indicate that the other protective relays are bypassed during LOCA conditions.

(1) Please clarify whether these other protective relays are bypassed only during LOCA or whether they are bypassed during both LOCA and LOPP conditions.

(2) Also verify that the diesel generator protective trips meet the other criteria specified in position C.7 and C.8 of Regulatory Guide 1.9, Rev. 2 (i.e., that they include the capability for (1) testing the status and operability of the bypass circuits, (2) alarming in the control room abnormal values of all bypass parameters, and (3) manually resetting of the trip bypass function (automatic reset not acceptable), and the surveillance system indicates which of the diesel generator protective trips is activated first).

435.14

Section 8.3.1.1.7 states that, in general, non-Class 1E loads are tripped off and thereby automatically isolated from the Class 1E buses by a LOCA or LOPP signal. Please verify that LOCA and LOPP signals are used to trip non-Class 1E loads and the loads are not subsequently resequenced back on automatically.

435.15

(1) Section 8.3.1.1.7(1) states that should the Class 1E bus voltage decay to below 70% of its nominal rated value for a predetermined time a bus transfer is initiated and the signal will trip the supply breaker, and start the diesel generator. Please provide the value of “predetermined time” (time delay) associated with bus voltage below 70%.

(2) Also, the last sentence in this section states that large motor loads will be sequence started as required and as shown on Table 8.3-2. Table 8.3-2, however, is only a “D/G Load Table” that does not identify any load sequencing times. Table 8.3-4 on the other hand is entitled “Load Sequence”, but the Table is “to be provided by December 31, 1988.” Please identify the correct table that will contain load sequencing times.

435.16

Section 8.3.1.1.7(2) states that if bus voltage (normal preferred power) is lost during post-accident operation, transfer to diesel generator power occurs as described in (1) above describes the normal sequence of operations following a LOPP). This, however, does not fully describe all the sequence of operations that need to occur for a LOCA followed by a LOPP.

(1) If the LOPP occurs near the beginning of the LOCA sequence before the diesel generator has accelerated to full speed and voltage on standby what occurs?

(2) If the LOPP occurs in the middle of the LOCA sequence after the diesel generator has accelerated to full speed and voltage on standby what occurs?

(3) If the LOPP occurs following completion of LOCA sequencing with diesel running in standby at full voltage and frequency what occurs?

(4) How is residual voltage handled when making the transfer from preferred power to the diesel generator with the diesel generator running in standby?

(5) Are non-Class 1E loads sequenced onto the diesel generator when the LOPP follows a LOCA?

The LOPP following LOCA sequence is important because, if a LOPP occurs as a result of a LOCA and the subsequent trip of the main generator, it may likely happen several seconds after the LOCA due to a sequence of events resulting in an unstable or overloading grid.

435.17

Section 8.3.1.1.7 does not have a scenario addressing the sequence of events that occur for a LOCA without a LOPP. Please address this scenario and add it to Section 8.3.1.1.7. If LOCA loads are sequenced on to the offsite power system, the sequencer used should be separate from that used to sequence loads on to the onsite power system. If this is not the case provide a detailed analysis to demonstrate that there are no credible sneak circuits or common failure

modes in the sequencer design that could render both onsite con offsite power sources unavailable. In addition provide information concerning the reliability of your sequencer and reference design detailed drawings.

435-18

Section 8.3.1.1.7(3) addresses the LOCA following LOPP scenario, however it provides few details.

(1) If the LOCA occurs just after the LOPP but prior to load sequencing of the LOPP loads what occurs?

(2) If the LOCA occurs in the middle of the LOPP sequence, what occurs?

(3) If the LOCA occurs following completion of the LOPP sequence, what occurs?

(4) Are any LOCA loads not already energized simply sequenced on to whatever LOPP loads are on-line or are some or all of the LOPP loads load-shed first?

(5) Are non-Class IE loads tripped by the LOPP signal or the LOCA signal?

(6) Is the diesel generator circuit breaker tripped at any time to accomplish the LOCA following LOPP response?

435.19

Section 8.3.1.1.7(4) states that if a LOCA occurs when the diesel generator is paralleled with the preferred power source during test and the test is being conducted from the local control panel, control must be returned to the main control room or the test operator must trip the diesel generator breaker. Because the diesel generator is not available to automatically respond to the LOCA in this circumstance it is considered to be bypassed and automatic indication of the bypass should be provided in the control room in accordance with Regulatory Guide 1.47. Please verify that this is the case.

435.20

In section 8.3.1.1.7(5) the description of what occurs following a LOPP during a diesel generator paralleling test with the normal preferred power source is different from that described for a paralleling test with the alternate preferred power source. In the first case it is stated that the diesel generator circuit breaker is automatically tripped if the normal preferred power supply is lost during the test, and in the second case it is stated that the diesel generator breaker will trip on overcurrent if the alternate preferred source is lost during the test.

(1) If what occurs during the two scenarios are different describe the differences and why they are different.

(2) If the diesel generator breaker is automatically tripped identify what signal will trip it since an undervoltage condition may not be generated.

(3) If the diesel generator breaker is tripped on overcurrent verify that no lock-outs will be generated to preclude automatic sequencing of LOPP loads.

(4) Verify that in either case the diesel generator will be returned to the isochronous mode prior to load sequencing.

(5) Describe what happens if a diesel generator bus fault occurs during the paralleling test.

435.21

(1) Section 8.3.1.1.8.2 is entitled “Ratings and Capability” but it provides no diesel generator ratings. Please provide the continuous load rating and short time overload rating of the diesel generators.

(2) In addition this section states that each diesel generator is capable of reaching full speed and voltage within 13 seconds after the signal to start. Does the diesel generator contain a ramp generator or some other circuitry to provide a controlled acceleration to operating speed during this 13 second starting period? If so, how will the reliability of this circuit be demonstrated?

435.22

Section 8.3.1.1.8.5 lists the diesel engine and its generator breaker protective trips and other off-normal conditions that are annunciated in the main control room and/or locally. Please identify which of these conditions are annunciated in the main control room and which are annunciated locally.

With regard to the diesel generator alarms in the control room: A review of malfunction reports of diesel generators at operating nuclear power plants has uncovered that in some cases the information available to the control room operator to indicate the operational status of the diesel generator may be imprecise and could lead to misinterpretation. This can be caused by the sharing of a single annunciator station to alarm conditions that render a diesel generator unable to respond to an automatic emergency start signal and to also alarm abnormal, but not disabling, conditions. Another cause can be the use of wording of an annunciator window that does not specifically say that a diesel generator is inoperable (i.e., unable at the time to respond to an automatic emergency start signal) when in fact it is inoperable for that reason.

Review and evaluate the alarm and control circuitry for the diesel generators in the ABWR design to determine how each condition that renders a diesel generator unable to respond to an automatic emergency start signal is alarmed in the control room. These conditions include not only the trips that lock out the diesel generator start and require manual reset, but also control switch or mode switch positions that block automatic start, loss of control voltage, insufficient starting air pressure or battery voltage, etc. This review should consider all aspects of possible diesel generator operational conditions, for example test conditions and operation from local

control stations. One area of particular concern is the unrest condition following a manual stop at the local station which terminates a diesel generator test and prior to resetting the diesel generator controls for enabling subsequent automatic operation.

Provide the details of your evaluation, the results and conclusions, and a tabulation of the following information:

(1) All conditions that render the diesel generator incapable of responding to an automatic emergency start signal for each operating mode as described above;

(2) the wording on the annunciator window in the control room that is alarmed for each of the conditions identified in (a);

(3) any other alarm signals not included in (a) above that also cause the same annunciator to alarm;

(4) any condition that renders the diesel generator incapable of responding to an automatic emergency start signal which is not alarmed in the control room; and

(5) any proposed modifications resulting from this evaluation.

For additional information and the staff position on this item see Branch Technical Position (BTP) PSB-2 in the Standard Review Plan (NUREG-0800). Describe how the ABWR design meets each position of BTP PSB-2.

435.23

Section 8.3.1.2.1 states that there are four 6.9kV electrical divisions, three of which are independent load groups backed by individual diesel generator sets. Figure 8.3-2 entitled “6.9 kV System Single Line” however shows only the three divisions backed by diesel generators. It does not show the fourth 6.9 kV division referred to in section 8.3.1.2.1. Please clarify this discrepancy and show the fourth division, if it exists, in Figure 8.3-1 and 8.3-2.

435.24

In section 8.3.1.2.1 it is stated that the standby power system redundancy is based on the capability of any two of the four divisions (two of three load groups) to provide the minimum safety functions necessary to shut down the unit in case of an accident and maintain it in the safe shutdown condition. Why can’t the unit be shut down in case of an accident with only one of the three load groups available? Identify the systems or loads needed that require that two of the three load groups be available.

435.25

In sections 8.1.3.1.2.3(6) and 8.3.1.2.1(3) it is stated that the undervoltage detection schemes for the 6.9 kV offsite power feeders is outside the nuclear island scope of supply, and BTP PSB-1 is therefore imposed as an interface requirement for the applicant. On the contrary however, the purpose of the undervoltage protection logic required by the BTP is to protect and ensure

the adequate operation of safety equipment at the 6.9 kV safety buses and below. It is required to be qualified Class 1E and should be physically located at and electrically connected to the Class 1E 6.9 kV switchgear. The undervoltage protection logic therefore protects equipment that is within the nuclear island scope, monitors voltage on the 6.9 kV safety buses that are within the nuclear island scope, and should be located in Class 1E 6.9 kV switchgear that is within the nuclear island scope. The setpoints of the undervoltage relays should be chosen to protect and ensure adequate operation of all safety loads down to the 120 volt level. The only connection between the requirements of the undervoltage protection and the 6.9 kV offsite feeders is that the feeders should be required to maintain adequate voltages to the safety buses under all operating conditions to ensure acceptable operation of safety equipment and to ensure that the undervoltage relays will not be unintentionally tripped. This should be accomplished by imposing appropriate interface requirements on the offsite feeders. You should therefore provide the second level undervoltage protection required by the BTP and address the other positions of the BTP PSB-1.

435.26

Clarify statement (1)(b) of section 8.3.1.2.2 regarding conformance of the SSLC power supply to GDC 2, 4, 17, and 18. If the SSLC power supply is not in conformance with any part of the GDCs so state and justify.

435.27

In section 8.3.1.2.2 states that the SSLC redundancy is based on the capability of any two of the four divisions to provide the minimum safety functions necessary to shut down the unit in case of an accident and maintain it in the safe shutdown condition. Why can’t the unit be shut down in case of an accident with only one of the four divisions available? Identify the systems or loads needed that require that two of the four divisions be available.

435.28

In section 8.3.1.2.4, item (1) states that certified proof tests are performed on cable samples to certify 60 year life by thermal aging. Subsequent items, (2) through (5), identify various cable attributes such as radiation resistance, mechanical/electrical endurance, flame resistance, and level of gas evolution that are also demonstrated by certified proof tests performed on cable samples. Do the tests identified in items (2) through (5) demonstrate that the cables have an acceptable level of the particular attributes at the end of their 60 year life? How is this demonstrated?

435.29

(1) Section 8.3.1.3.1 discusses the means used to physically identify safety related power systems equipment. It states that all cables for Class 1E systems and associated circuits (except those routed in conduit) are tagged every 15 ft. In addition all cables are tagged at their terminations with a unique indentifying number. Regulatory Guide 1.75, Rev. 2 states that these cables should be marked at intervals not to exceed 5 ft. and the preferred method of marking the cable is color coding. IEEE-384-1974 also

states that these cable markings shall be applied prior to or during installation. Please verify that these recommendations are met or justify the differences. If exception is taken to position C.10 of Regulatory Guide 1.75, Rev. 2 regarding cable marking, the exception should be identified in section 8.1.3.1.2.2 and wherever the exception is applicable.

(2) Section 8.3.1.3.1 also describes the marking of conduit and cable trays. Please verify that in accordance with the requirements of IEEE-384-1974 these markings are applied prior to the installation of cables.

(3) The identification requirements for instrumentation and control system cables and raceways described in items (3) and (4) of section 8.3.1.3.2.1 should be the same as those for power systems provided in section 8.3.1.3.1 subject to the above comments.

435.30

Provide a description of the ABWR cable spreading areas in the ABWR SSAR. Describe how the requirements specified in section 5.1.3 of IEEE-384-1974 (as modified by position C.12 of Regulatory Guide 1.75) are met.

435.31

(1) Item (7) of Section 8.3.1.4.1.2 discusses electric penetration assemblies. It states that electric penetration assemblies of different Class 1E divisions are separated by distance, separate rooms or barriers and/or locations on separate floor levels. With regard to separation by distance, no specifics are given on what is the minimum distance provided between redundant penetrations. As required in IEEE-384-1974 the minimum physical separation for redundant penetrations should meet the requirements for cables and raceways given in section 5.1.4 of that standard. Please verify that this is the case.

(2) Item (7) of section 8.3.1.4.1.2 also states that power circuits going through electric penetration assemblies are protected against overcurrent by redundant overcurrent interrupting devices to avoid penetration damage. The use of redundant overcurrent interrupting devices should not be limited to only power circuits going through electric penetration assemblies. They should be used on all penetration electric circuits (including instrumentation and control circuits) where the available fault current is greater than the continuous rating of the penetration, but is greater than the continuous rating of a device upstream of the penetration whose failure can result in fault current levels in excess of the penetration continuous rating (such as a control power transformer), then redundant overcurrent interrupting devices should be used. Please verify that this is the case.

(3) Provide the fault current clearing-time curves of the electrical penetrations' primary and secondary current interrupting devices plotted against the thermal capability (I^2t) curve of the penetration (to maintain mechanical integrity). Provide a simplified one-line diagram on this drawing showing the location of the protective devices in the penetration circuit, and indicate the maximum available fault current of the circuit.

(4) Where external control power is needed for tripping electrical penetration breakers, signals for tripping the primary and backup breakers should be independent, physically separated and powered from separate sources. Verify that your design complies and identify the power supplies to the redundant circuit breakers.

435.32

Section 8.3.1.4.2.1 identifies the standards that are used for the separation of equipment for the systems referred to in subsection 7.1.1.3, 7.1.1.4, and 7.1.1.6 (safety-related control and instrumentation systems). IEEE-384-1974 however is not listed. The separation of equipment in these systems should comply with the requirements of this standard. Please verify that this is the case. In addition, the listed standards and requirements are not identified as being applicable to subsection 7.1.1.5 (safety-related display instrumentation). Please verify that they are indeed applicable to this subsection.

435.33

Items (4) and (5) in section 8.3.1.4.2.2.2 state that spatial separation in general plant areas and in cable spreading areas shall equal or exceed the minimum allowed by IEEE-384. IEEE-384-1974 however provides two means for establishing minimum physical separation distances. The first, which is specified in section 5.1.1.2 of the standard allows the minimum separation distance to be established by analysis based on tests of the proposed cable installation. The second, which is specified in sections 5.1.3 and 5.1.4 of the standard, specifies specific minimum physical separation distances that must be maintained. Please clarify weather you intend to meet the specific distances specified in the standard or weather you intend to establish your own separation distances through analysis based on tests. The preferable option is to meet the specific distances specified in IEEE-384-1974.

435.34

(1) Section 8.3.1.4.2.2.4 discusses the use of isolation devices in power circuits. It states that non-Class 1E instrument and control circuits will not be energized from a Class 1E power supply unless potential for degradation of the Class IE power source can be demonstrated to be negligible by effective current or voltage limiting (i.e., functional isolation) under all design basis conditions. Please explain what this means. Does it imply that no isolation device will be used if no credible failure modes can be identified that will result in fault currents? Qualified isolation devices should be used in all cases where a non-Class 1E circuit is connected to a Class 1E power supply.

(2) It also states in section 8.3.1.4.2.2.4 that Class 1E power supplies which interface non-Class 1E circuits are required to be disconnected or otherwise decoupled from the non-Class 1E circuits such that conditions of the non-Class IE portions (e.g., by current limiting element). Verify that, if overcurrent interrupting devices such as fuses or circuit breakers are used as isolations where there is an interface between a Class 1E power supply and non-Class 1E circuit. Identify the isolation device that is used at the interface.

(3) Where redundant Class 1E power circuits interface with a common non-Class 1E system such as a computer, the isolation devices used should ensure that a worst case abnormal occurrence (fault, overvoltage, voltage surge or spike, etc.) on one of the Class 1E power circuits cannot migrate through the non-Class 1E system and affect the redundant Class 1E circuit. This is in addition to the normal criteria for isolation devices that require that any worse case occurrence (maximum credible faults, etc.) in the non-Class IE system not affect the Class 1E system.

435.35

Item (4) of section 8.3.1.4.2.3.1 states that the scram solenoid conduits will have unique identification but no specific requirements, and the scram group conduits may run in the same raceway with other divisional circuits. If the scram group conduits are run in the same raceway with other divisional circuits or if they have less than the minimum separation from Class 1E circuits, they must be treated as associated circuits and must meet the requirements specified in section 4.5 of IEEE-384-1974. Please verify that this is the case, and identify the specific separation requirements that will be applied to the scram group conduits when they become associated circuits.

435.36

Item (6) of section 8.3.1.4.2.3.2 states that any electrical equipment and/or raceways for RPS or ESF located in the suppression pool level swell zone will be designed to satisfactorily complete their function before being rendered inoperable due to exposure to the environment created by the level swell phenomena. This information is not sufficient for us to evaluate the effects on flooding of electrical equipment. Please identify all electrical equipment, both safety and non-safety, that may become submerged as a result of the suppression pool level swell phenomena or as result of a LOCA. For all such equipment that is not qualified for service in such an environment provide an analysis to determine the following:

(1) The safety significance of the failure of this equipment (e.g., spurious actuation or loss of actuation function) as a result of flooding

(2) The effects on Class 1E electrical power sources serving this equipment as a result of such submergence

(3) Any proposed design changes resulting from this analysis

435.37

In the description of the DC power system in section 8.3.2.1 it is stated that the operating voltage range of Class 1E DC loads is 105 to 140 V. It is also stated that maximum equalizing charge voltage for Class 1E batteries is 140 VDC, and the DC system minimum discharge voltage at the end of the discharge period is 1.75 VDC per cell. For a 125 VDC lead acid battery with 60 cells. 1.75 VDC per cell equates to a final discharge voltage of 105 VDC at the battery terminals. This is the same as the stated minimum operating voltage of the Class 1E DC loads. There is therefore no allowance for voltage drop from the battery terminals to the terminals of the Class 1E loads at the final voltage value of 1.75 VDC per cell. Please address this discrepancy. Also, provide the results of your DC voltage analysis showing battery terminal voltage and worst case DC load terminal voltage at each step of the Class 1E battery loading profile. See the following question with regard to the battery loading profile.

435.38

Section 8.3.2.1 addresses the DC power systems in general and section 8.3.2.1.3.2 specifically addresses battery capacity. With regard to battery capacity, section 8.3.2.1.3.2 states that battery capacity is sufficient to satisfy a safety load demand profile under the conditions of a LOCA and loss of preferred power, the batteries have sufficient stored energy to operate connected essential loads continuously for at least two hours without recharging.

(1) Provide the stated load demand profiles and a breakdown of the loading during this demand.

(2) Provide the manufacturer's ampere-hour rating of the batteries at the two hour rate and at the eight hour rate, and provide the one minute ampere rating of the barriers.

(3) Address station blackout with regard to battery capacity. If a station blackout coping analysis is being prepared for the ABWR, provide a battery load demand profile for the coping duration. Provide a breakdown of the loading during this demand.

435.39

In section 8.3.2.1 it is stated that each 125 VDC battery is provided with a charger and a standby charger shared by two divisions, each of which is capable of recharging its battery from a discharged state to a fully charged state while handling the normal, steady-state DC load.

(1) Provide the continuous and current-limited output ratings of the battery chargers.

(2) In accordance with position C.1.b of Regulatory Guide 1.32, Rev 2 verify that the capacity of the battery charger supply is based on the largest combined demands of the various steady-state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the charge status of the plant during which these demands occur.

(3) Verify that the battery charger can operate stably as a battery eliminator (i.e., with the charger remaining connected to supply the loads while the battery is disconnected from the loads).

(4) Verify that no reverse DC current can flow into the battery charger output from the battery, during periods of low AC input battery charger voltage or during total loss of low AC input voltage to the charger.

435.40

Section 8.3.2.1 and Figure 8.3-8 identify the connection of the non-Class 1E 250 VDC battery chargers to divisions I and 3 of the Class 1E system. Identify the isolation devices used at this interface. Are the Class 1E breakers shown at the interface, tripped on an accident signal? If not, they should be, or else redundant qualified breakers should be provided.

435.41

Section 8.3.2.1.2 very generally identifies the type of loads fed from the 125 VDC Class 1E power system. Please provide a more specific breakdown of the loads fed from each division of the 125 VDC Class 1E power system.

435.42

In Section 8.3.2.1.3 it is stated that an emergency eyewash is installed in each battery room. In order to ensure that water cannot be inadvertently splashed on the batteries the eyewash stations should be located away from the batteries and the eyewash installation and its piping should be seismically qualified. Please verify that this is the case.

435.43

Section 8.3.2.1.3.3 states that battery rooms are ventilated to remove the minor amounts of gas produced during charging of batteries. Verify that, in accordance with position C.1 of Regulatory Guide 1.128 the ventilation system will limit hydrogen concentration to less than two percent by volume at any location within the battery area. Also, in accordance with position C.6.e of Regulatory Guide 1.1.28, verify that ventilation air flow sensors are installed in the battery rooms with their associated alarms installed in the control room.

435.44

With regard to the DC power systems, section 8.3.2.2.1 states that all abnormal conditions of important system parameters such as charger failure or low bus voltage are annunciated in the main control room and/or locally. Please identify the specific meters and alarms used for monitoring the status of the Class 1E DC power systems and indicate weather they are located in the main control room and/or locally. As a minimum the following indications and alarms should be provided in the control room:

- Battery current (ammeter-charge/discharge)
- Battery charger output current (ammeter)

- DC bus voltage (voltmeter)
- Battery charger output voltage (voltmeter)
- Battery discharge alarm
- DC bus undervoltage and overvoltage alarm
- DC bus ground alarm (for ungrounded system)
- Battery breaker open alarm
- Battery charger trouble alarm (one alarm for a number of abnormal conditions which are usually indicated locally)

Because the ABWR is an advanced reactor design, you should consider the use of state-of-the-art art battery and electrical system monitoring system to assure immediate notification of battery and electrical system problems and to provide for the monitoring of at least the individual cell parameters of the batteries and the status of the various electrical system circuits, and ideally should provide for monitoring the status of all AC and DC system circuits down to and including all control circuits.

435.45

Section 8.3.3.1 states that conductors are specified to continue to operate at 100% relative humidity with a service life expectancy of 40 years. The following sentence states however that the Class 1E cables are designed to survive the LOCA ambient condition at the end of the 60-yr. life span. If the intent is to qualify the cables for the 60-year life of the plant, why is a service life expectancy of only 40 years specified for the 100% relative humidity condition?

435.46

The following questions pertain to Table 8.3-1 “D/G Load Table-LOCA,” Table 8.3-2 “D/G Load Table-LOPP.” and Table “Notes for Tables 8.3-1 and 8.3-2:”

(1) Please provide a translation for the acronyms used in these tables.

(2) Please correct the numerous errors/discrepancies between tables 8.3-1 and 8.3-2 regarding the ratings of the loads. There are many instances where the rating of an identical piece of equipment is different in table 8.3-1 from that given in table 8.3-2.

(3) Please explain why the loads shown on the diesel engine are larger than their rated values. If this is to account for losses through the generator please explain the advantage of calculating the loads on the diesel engine versus the more commonly used means of calculating the loads on the output of the diesel’s generator. Provide the factors and their rationale used for increasing the various loads from their rated values, since the loads are not all increased a like amount.

(4) Provide a more complete breakdown of the loads identified in the category “Other Loads”.

(5) Why is the load identified as “NPSS CVCF” listed as 31.8kW for the D/G “c” LOCA load while it is listed as 37.9kW for the D/G “c” LOPP load? In all other cases LOCA and LOPP loads are the same value if they are energized under both conditions.

(6) I do not understand note (5). It says, “Division III HPCF pump motor starts by L2 signal on the case of loss of preferred power (LOPP).” Table 8.3-2 however shows the HPCF pumps running on both divisions II and III (b and c) during a LOPP. Do not both motors start and run during a LOPP? Note (5) also says “As HPCF pump motors has very large capacity, they are connected to Div. II, III to equalize the DG load capacity.” What is the intent of this note? If the HPCF pumps are 100% redundant pumps, wouldn’t you want to connect their motors to different divisions anyway to preserve their redundancy?

(7) Note (6) states that the CUW pump may operate under LOPP condition, but not operate with SLC pump operation. On this calculation, it states, CUW pump is not considered because SLC pump is included. Because the CUW pump operating load is greater than the SLC pump operating load, the CUW pump load should be used instead of the SLC pump load during LOPP, in order to provide the worst case loading on the diesel generator. Please justify or change the table accordingly.

(8) Note (7) states that the TCW/TSW pumps are connected to non-div. switchgears. Although these pumps are listed in tables 8.3-1 and 8.3-2, no loading on the diesels are identified for these pumps. If these pumps cannot be connected to the diesel generators why are they shown in Tables 8.301 and 8.3-2? If they can be connected to the diesels, then a load should be identified for them on the diesels during the LOPP condition. This will provide worst case loading on the diesels during LOPP.

(9) Note (9) states that the remainder of plant equipment are connected to Div. I and if A and B motors are provided, they are connected to Div. I and II respectively. According to this note loads should only be shown on D/Gs “A” and “B” in the category (“Other Loads”) that the note refers to. There is, however, a load of 2l0kW shown on D/G “C” under this category. Please clarify this apparent discrepancy.

(10) Note (10) says, “Only part of HNCW) HVAC normal cooling water system) will be considered under LOCA case.” This note, however, is provided in the LOPP table (table 8.3-2). A note (note 3) is provided in the LOCA table (table 8.3-1) for this equipment which states, “Loads are shed with LOCA signal.” It appears then that note (10) should read, “Only part of HNCW) HVAC normal cooling water system) will be considered under LOPP case.” Please clarify weather this is the case. If the

foregoing is the case, a load for the HNCW equipment should be shown on the diesels for the LOPP condition (Table 8.3-2). Presently, a load on the diesel generators during LOPP is not identified for this equipment.

435.47

The following questions pertain to Figure 8.3-1 “Power Distribution Single Line Diagram”:

(1) The division II 6.9 kV bus is shown broken into two separate buses. This is apparently an error. Please correct.

(2) The circuit between the division III 6.9 kV bus and the 480V switchgear P/C 6E-1 does not show an intervening transformer. Please correct.

(3) Identify the ratings of the diesel generators and 6900/480V transformers on this drawing.

(4) Discuss the circuit from the division 1, II, and Ill 480V switchgear to the turbine island labeled as “To 480V Switchgear (Alternate Preferred Power).” If this is a power feed to loads in the turbine island identify the loads it feeds, the circumstances under which the loads are fed, and describe the IE/non-IE isolation provided. If this is a power feed from the turbine island identify the source of power and the need for a second source of power to the 480V Class 1E bus. In either case identify the interface requirements for this circuit.

(5) On every bus shown in Figure 8.3-1 there is one circuit shown connected to ground through a circuit breaker. Describe the function of this circuit. If the circuit is used to provide a safety ground on the bus during maintenance operations describe the interlocks, controls, and alarms provided to assure it is not inadvertently energized during non-maintenance operations.

(6) Note 2 on this drawing says, “See 480V MCC one-line diagram for details.” There is, however, no “480V MCC one-line diagram” provided in the SSAR. Please provide us this diagram and include it in the ABWR SSAR.

(7) The arrangement of the normal preferred and alternate preferred power sources to the 6.9 kV buses does not agree with that shown on Figure 8.3-2. Please correct this discrepancy.

435.48

The offsite power circuits to the 6.9 kV Class 1E buses shown in Figure 8.3-2 “6.9 kV System Single Line” should be appropriately labeled as “Normal Preferred Power” or “Alternate Preferred Power.” Also, the way the offsite circuits are arranged on this drawing makes it appear that they are connected to the same 6.9 kV High Voltage Switchgear as the RIPS. The offsite circuits to the Class 1E buses should be directly connected to a winding of the Offsite Power Transformers that is separate from that which feeds the non-Class 1E loads. The Offsite

Power Transformers, however, should have the capability of feeding both Class 1E and non-Class 1E loads so the plant does not have to rely on only Class 1E loads when only one offsite power source is lost. Also, the offsite power supply circuits to the Class 1E buses should be arranged so that all three Class 1E divisions are not simultaneously deenergized on the loss of only one of the offsite power supplies. These should be included as interface requirements. Please verify that this is the case.

435.49

With regard to Figure 8.3-3 “480V System Single Line”:

(1) Identify the feeds to 480V switchgear P/C 6A-1, P/C 6A-2, P/C 6B-1, and P/C 6B-2. Describe the purpose and function of these switchgear and the R/B MCCs they feed. Identify the type of loads they feed.

(2) Identify the location, purpose and function of P/C 6SB-1. Identify the type of loads it feeds. Why does it have feeds from all three divisions of 480V switchgear? Identify the isolation devices used, and provide a connection diagram of the three divisional feeds to P/C 6SB-1. If P/C 6SB-1 is outside the nuclear island provide its interface requirements.

(3) If the T/B MCCs are non-Class 1E identify the isolation devices used and the interface requirements.

435-50

The non-safety-related instrument power system shown in Figure 8.3-4 has two redundant Class 1E power feeds to it. Identify the isolation devices used between the Class 1E and non-Class IE systems. A Class 1E circuit breaker tripped on a LOCA signal or two redundant Class 1E circuit breakers coordinated with the upstream MCC feeder breaker are acceptable isolation devices.

435.51

On Figures 8.3-5, 8.3-6, 8.3-7, and 8.3-8 describe the function and operation of the various devices that are identified by device numbers. Also, on Figure 8.3-7 and 8.3-8 define the acronym SID located next to the diode device. Describe the function and operation of this device.

435.52

On Figure 8.3-7 “125 VDC Power System” describe the function and operation of the various key interlocks shown on the figure.

435.53

On Figure 8.3-8 “250 VDC Power System” describe the type of isolation provided between the Class IE divisional power feeds and the non-Class 1E DC Power System. Also describe the type

of isolation and separation provided between the power feed from P/C 6E-1 (Division Ill) and the power feed from P/C 6C-1.

435.55

Section 8.3.1.1.8.9 states that the qualification tests are performed on the diesel generator per IEEE-387 as modified by Regulatory Guide 1.9 requirements. If the qualification tests have been performed please provide us the results of the tests. If the tests have not yet been performed please indicate at what point the tests will be conducted.

435.56

There have recently been a number of problems identified with the electrical systems at Nuclear Power Plants. Although a number of these arose as a result of modifications done on the electrical systems after the plants were licensed, some were or could have been the result of poor original design.

(1) Generic Letter 88-15 addresses a number of electrical system problems that have occurred primarily as a result of inadequate control over the design process. Some of these inadequacies have occurred in areas of electrical system design which have historically well established principles such as circuit breaker coordination and fault current interruption capability. As a result the staff has not normally undertaken a detailed review of these areas, relying instead on the designers exercise of these well established principles. It is important that these areas have comprehensive, detailed design criteria and guidelines established for the design engineer. Controls should exist to ensure that these criteria are followed during the design process. Please address the specific problems discussed in GL 88-15 identifying the criteria and guidelines used to ensure that these inadequacies will not be found in the ABWR design. Provide a general discussion of the controls that exist over the design process in the electrical system area of the ABWR design.

(2) RC Bulletin No. 88-10 and NRC Information Notice No. 88-46 identifies a problem with defective refurbished circuit breakers. Although the primary concern is with circuit breakers used in safety-related circuits, there is also a concern with non-safety-related breakers used for electrical penetration protection, since these also provide a safety-related function but undergo less scrutiny. Please identify how you ensure that non-Class 1E breakers purchased for use in containment electrical penetration circuits are high quality, new circuit breakers from the circuit breaker manufacturer, rather than refurbished circuit breakers.

435.57

With respect to the application of single failure criterion to manually-controlled, electrically-operated valves, list all valves for which SRP Branch Technical Position ICSB 18 (PSB) may apply. Describe (1) how power is locked out to active and passive valves, (2) how power can be reinstated from the control room if valve repositioning (active valves) is required later, and (3) how the valve position indication meets the single failure criterion.

435.58

Experience with Nuclear Power Plant Class 1E electrical equipment protective relay applications has established that relay trip setpoint drifts with conventional type delays have resulted in premature trips of redundant safety related system pump motors when safety system was required to be operative. While the basic need for proper protection for feeders/equipment against permanent faults is recognized, it is the staff’s position that total non-availability of redundant safety systems due to spurious trips in protective relays is not acceptable. Provide a description of your circuit protection criteria for safety systems/equipment to avoid incorrect initial setpoint selection and the above cited protective relay trip setpoint drift problems.

435.59

Explicitly identify all non-Class 1E electrical loads which are or may be powered from the Class 1E AC and DC systems. For each load identified provide the horsepower or kilowatt rating for that load and identify the corresponding bus number and division from which the load is powered. Also identify the type of isolation device used between the non-Class 1E load and Class 1E power supply.

435.60

Section 8.3.1.2.1 states compliance with the recommendations of Regulatory Guide 1.106 “thermal Overload Protection for Electric Motors on Motor-Operated Valves.” Describe the means used to bypass the thermal overload protection to Class 1E MOVs during accident conditions. Describe what type of indication for the bypass or lack of bypass is provided in the control room. Provide a schematic of the design or give MOV drawing references as specific examples of the design.

435.61

Experience with nuclear power plant Class 1E motor-operated valve motors has shown that in some instances the motor winding on the valve operator could fail when the valve is subjected to frequent cycling. This is primarily due to the limited duty cycle of the motor. Provide the required duty cycle of the ECCS and RCIC steam and water line motor operated valves as they relate to their respective system modes of operation during various events. Demonstrate that the availability of the safety systems in the ABWR design will not be compromised due to the limited duty cycle of the valve operator motors.

435.62

Provide the minimum required starting voltages for Class 1E motors. Compare these minimum required voltages to the voltages that will be supplied at the motor terminals during the starting transient when operating on offsite power and when operating on the diesel generators.

20.2.9 Chapter 9 Questions

281.11

In Section 9.1.3.1.1, conductivity units of umhos/cm are used while in Section 9.2.9.1 units of uS/cm are used. These units should be consistent. (9.1.3)

281.12

In Section 9.1.3.1.1, the pH range of 5.6 to 8.6 should be at 25°C.

410.31

Identify, in detail, the principal equipment that comprise the spent fuel pool cooling and cleanup system. Be specific to define its boundaries and safety-related portions. (9.1.3)

410.32

Explain why, for the ABWR, two 50% rated spent fuel pool cooling trains are considered as sufficient (Note that for some of the operating BWRs, two 100% rated spent fuel pool cooling trains have been provided). (9.1.3)

410.33

Explain why, for the ABWR, the minimum capacity of the spent fuel pool for storage of the spent fuel is only 270% of a full core. (9.1.3)

410.34

In the design bases for the Fuel Pool Cooling and Cleanup System (FPCS), include the requirements for makeup water and radiation shielding. Provide appropriate discussion regarding the design compliance with Regulatory Position C.8 of Regulatory Guide 1.13 and Regulatory Positions C.2.f(2) and C.2.f(3) of Regulatory Guide 8.8. (9.1.3)

410.35

Discuss the extent of the system's compliance with Regulatory Positions (9.1.3) C.1, C.2, and C.6 of Regulatory Guide 1.13, Regulatory Positions C.1 and C.2 of Regulatory Guide 1.29, and address the Quality Group requirements for the system in accordance with Regulatory Guide 1.26.

410.36

With respect to the cooling capacity of the system, demonstrate that on the bases of conservative assumptions relative to Branch Technical Position 9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling," and SRP 9.1.3 III.1.h, the total capacity of the heat exchangers with both pumps operating exceeds the maximum normal heat load, and that the fuel pool temperature can be maintained below the 140°F criterion, specified in SRP Section 9.1.3 for maximum normal conditions. Also, confirm that 140°F will not be exceeded if a single active failure and loss of offsite power is assumed. Describe the redundancy provisions in powering the two cooling pump motors. (9.1.3)

410.37
Describe the emergency makeup water systems provided, and discuss redundancy and seismic requirements for the system. (9.1.3)

410.38
Discuss control-room-alarmed fuel pool water temperature, fuel pool water level and building radiation level monitoring systems provided to satisfy GDC 63. (9.1.3)

410.39
Discuss provisions for, or provide descriptions of, the following design features: (1) leakage detection system, (2) individual isolation capabilities for components and headers to assure system leakage control and maintenance, (3) capability to detect radioactivity and/or chemical contamination transfer from one system to another, and (4) protection of the various components of the fuel pool cooling system against failures of other applicable moderate and high-energy piping systems. (9.1.3)

410.40
A complete evaluation of the Overhead Heavy Load Handling System (OHLHS) cannot be performed without the descriptions of the new and spent fuel storage facilities to be provided in Section 9.1.1 and 9.1.2 of the ABWR SSAR. Also, Section 9.1.4 “Light Load Handling System” is needed, as it is extensively referenced in 9.1.5. All these Sections, 9.1.1, 9.1.2, and 9.1.4, are scheduled to be submitted to the NRC in December, 1988. Note that on completion of review of the above sections, additional information on Section 9.1.5 may be requested. (9.1.5)

410.41
Provide the seismic category, safety class, and quality group for all components used in the OHLHS and discuss the system design in terms of conformance with the regulatory positions of Regulatory Guides 1.13 and 1.29. Discuss the system design in terms of conformance with the guidelines of NUREGs-0554 and 0612 as they relate to protection against natural phenomena. (9.1.5)

410.42
Identify all the individual heavy load handling systems (names and hoist trolleys) that have been designed to meet the single-failure-proof requirements in accordance with the guidelines of NUREG-0554. Identify the safety factors provided for slings and strongbacks. Also provide the results of a failure mode and effects analysis demonstration that the individual subsystems and components including controls and interlocks are designed to meet the single-failure criterion without compromising the capability of the OHLHS to perform its safety function. (9.1.5)

410.43
Discuss compliance with GDC 4, “Environment and Missile Design Bases” and GDC 61, “Fuel Storage and Handling and Radioactivity Control” as it relates to handling the spent fuel cask. (9.1.5)

410.44
Provide P&IDs for the Condensate Storage Facilities and Distribution System (i.e., Makeup Water Condensate (MUWC) System). Also, provide a list of tanks (with capacity) and other requirements in the system. (9.2.9)

410.45
Clarify which portion of the MUWC system is within the ABWR scope. Also, identify the system interfaces which include flow rates, supply pressure and temperature. (9.2.9)

410.46
Clarify whether the distribution system includes any surge volume and, if so, how much and for suction of which pumps. Also, if applicable, describe how protection against the effects of flooding resulting from possible failure of the surge volume is ensured. Define what “HPCF pumps” means. (9.2.9)

410.47
Describe the design features provided in the system and/or interfacing components to ensure automatic switchover of the suction of the applicable pumps to safety-related water sources, if so required. (9.2.9)

410.48
Discuss conformance of the MUWC systems design with the requirements 10CFR50.63, “Loss of all Alternating Current Power.” Specifically include the system’s capacity and capability to ensure core cooling by removing decay heat independent of preferred and onsite emergency ac power in the event of a station blackout for the specified duration, in accordance with guidelines of Regulatory Guide 1.55, “Station Blackout,” Positions C.3.2 through C.3.5, as applicable. (9.2.9)

410.49
Discuss compliance of the system with Positions C1 and C2 of Regulatory Guide 1.29. (9.2.9)

410.50
Provide P&IDs for the Demineralized Water Makeup System (i.e., Makeup Water System (Purified) (MUWP)). (9.2.10)

410.51
Clarify which portion of the MUWP is within the ABWR scope. Also, identify the system interfaces which include temperature, chemistry, system capacity (i.e., tank volume) and treatment. (9.2.10)

410.52
Provide the water quality characteristics for the MUWP water (SSAR Section 9.2.10.1, Item 3, refers to Section 9.2.8 which in turn refers to Section 9.2.16. However, Section 9.2.16 does not give the water quality characteristics). (9.2.10)

410.53
Discuss compliance of the system with Position C1 (e.g., containment penetration portions) and Position C2 of Regulatory Guide 1.29. (9.2.10)

410.54
Verify that flooding analyses have been performed for a failure of the nonseismic Category I demineralized water makeup system where the piping runs through safety-related structures and tunnels containing safety-related equipment. (9.2.10)

410.55
With respect to the capability of the Reactor Building Cooling Water System for detection, control, and isolation of system leakage, and radioactive leakage: (9.2.11)

(1) Identify the isolation valves which isolate the non-essential loads from the essential supply headers and describe their isolation function in the event of a LOCA or in the event of a leak detected in the non-essential system piping.

(2) Identify and describe instrumentation used to detect leakage in the non-essential system piping.

(3) Identify the valves which are activated by the surge tank level switch to isolate a leaking system train.

(4) Identify all radiation monitors provided and describe their individual function. Also, clarify whether the system design includes any radiation monitor in the pump suction header to detect inleakage from radioactive systems.

410.56
Identify the functional performance requirements associated with water hammer and address the design provisions and procedures provided to meet these requirements. (9.2.11)

410.57
Identify the system requirements for water makeup, and address the capacity of the surge tanks to accommodate expected leakage from the system or that a seismic source of makeup water can be made available within a time frame consistent with surge tank capacity. (9.2.11)

410.58
Provide the design characteristics for the system pumps, tanks and heat exchangers. (9.2.11)

410.59

Define the terms: FCS, CAMS, LWC, HSCR, HWH hot water heat exchanger, and HCW. (9.2.11)

410.60

Discuss how the Reactor Building Cooling Water (RCW) system complies with Position C2 of Regulatory Guide 1.29 with respect to the non-safety related portions, and with respect to GDC 2 for safety-related portions (e.g. physical location to protect against earthquakes and floods). (9.2.11)

410.61

Clarify whether the flows indicated for the components serviced by the RCW system in SSAR Tables 9.2-4a, 4b and 4c represent the minimum flow requirements at the inlet of each component. Also, specify the maximum allowable RCW temperature at the inlet of each component under different operating conditions. (9.2.11)

410.62

Clarify whether availability of only one division of the RCW system is sufficient to provide cooling water to the drywell coolers and the RIP coolers (SSAR Tables 9.2-4a and 4b list only Division A and B servicing above. Further Table 9.2-4b lists only the Drywell B cooler as being serviced by Division B). (9.2.11)

410.63

Regarding the HVAC Normal Chilled Cooling Water System, provide information on the following: (9.2.11)

(1) Compliance with GDC 2 for safety-related components (i.e., physical location for complying with the GDC).

(2) Compliance with Position C2 of Regulatory Guide 1.29 for the non-safety related portion.

(3) Automatic features to provide cooling water to the equipment serviced by the system in the event of its failure on loss of offsite power (specify the system that will provide cooling water in the above situation).

(4) Description of the turbine building cooling water system which provides condenser cooling (refer to SSAR Section 9.2.12.2) if it is within the ABWR scope. Otherwise, identify it as an interface requirement.

410.64

Regarding the HVAC Emergency Chilled Cooling Water system, provide information on the following: (9.2.13)

(1) Compliance with GDC 2 for safety related portion (i.e., physical location for complying with the GDC).

(2) Compliance with Position C2 of Regulatory Guide 1.29 for the non-safety related portion, if there is any such portion.

(3) Compliance with GDC 4.

(4) System active component failure analysis.

420.107

Describe procedural controls considered adequate to control the keylocked SLCS. (9.3.5.2.1)

420.117

Describe interlocks and indications used to prevent injection of the testing mode demineralized water instead of boron. (9.3.5.1.1)

430.177

ABWR SSAR Section 9.1.1.1.1, Nuclear Design, states that since no credit is taken for neutron leakage, the value for effective multiplication factors are really infinite neutron multiplication factors. ABWR SSAR Section 9.1.13.1, Criticality Control, states that k_{eff} for both normal and abnormal storage conditions will be less than or equal to .95. However, the same section states that the new fuel storage area will accommodate fuel with a $k_{inf} < 135$ with no safety implications. Resolve this discrepancy. (9-1-1)

430.178

ABWR SSAR Section 9.1.1.1.6, Dynamic Analysis, refers to ABWR SSAR Section 9.1.2.1.6, which does not exist. Provide the results of a dynamic analysis of the new fuel storage system. (9.1.1)

430.179

ABWR SSAR Section 9.1.1.1.7, Impact Analysis, also refers to a nonexistent A.BWR SSAR Section 9.1.2.1.7. Provide impact analysis for “impact” loads up to and including a fuel assembly and its carrying fixture. (9.1.1)

430.180

Provide details of assumptions and input parameters used in the criticality analysis for new fuel storage. Include information such as number of racks, their material (e. g., stainless steel?), number of fuel assemblies per rack, neutron-absorbing material and its placement, placement of fuel assemblies (center-to-center distance between rows and within rows), and effect of spacing on k_{eff} in normal dry condition or when completely flooded with water. Also, clarify

whether the spacing is sufficient to ensure a k_{eff} of 0.98 or less under optimum moderator conditions (foam, small droplets, spray or fogging) as described in SRP Section 9.1.1. Clarify whether the racks are designed to preclude inadvertent placement of a fuel assembly in other than prescribed locations. (9.1.1)

430.181

How is the new fuel protected from internally generated missiles and the effects of moderate or high energy piping or rotating machinery in the vicinity of the vault housing the new fuel storage racks. (9.1.1)

430.182

Provide information on how the design of the new fuel storage facility complies with GDC 61, "Fuel Storage and Handling and Radioactivity Control." Identify the ventilation system provided to handle possible release of radioactivity resulting from accidental damage to the fuel (note that ABWR SSAR 7.1 does not describe the radiation monitoring equipment for the new fuel storage area as stated in ABWR SSAR Section 9.1.1.2). (9.1.1)

430.183

Provide sufficient information and drawings to determine that the failure of non-seismic systems and structures in the vicinity of the new fuel storage facility can not cause an unacceptable increase in k_{eff} (9.1.1)

430.184

Demonstrate that the analyzed impact of a fuel assembly, including its associated handling tool, dropped from a height of 6 feet bounds the range of all possible load drops from all possible heights. For additional guidance on the required bounding analysis, see SRP Section 9.1.2, Item III.2.e.(9.1.2)

430.185

Provide sufficient information and drawings to determine that the failure of non-seismic systems and structures in the vicinity of the spent fuel storage facility can not cause an unacceptable increase in k_{eff} (9.12)

430.186

Provide drawings and information pertaining to spent fuel transfer canal capability of the fuel transfer canal or other provisions to prevent a dropped shipping cask from causing an unacceptable loss of pool water.(9.12)

430.187

Clarify whether there is a) an interconnecting fuel transfer canal capable of being isolated from the fuel pool and adjacent cask loading area, and b) any high-energy piping or rotating machinery in the vicinity of the fuel storage pools. Also, clarify whether the racks are designed to preclude inadvertent placement of a fuel assembly in other than prescribed locations.(9.12)

430.188

Describe the function of the containment pool mentioned in ABWR SSAR Section 9.1.2.1.5. (9.1.2)

430.189

What is the seismic category of the gates in the pools? (9.1.2)

430.190

Instead of referring to a specific GE proprietary report on criticality control for spent fuel storage (see ABWR SSAR Section 9.1.2.3.1), provide details of assumptions and input parameters used in the criticality analysis of the spent fuel storage. Also provide the uncertainty value and associated probability and confidence level for the k_{eff} value determined by the analysis. Include information such as number of fuel assemblies stored in the pool, center-to-center spacing between fuel assemblies, material of the racks, neutron absorber used and its placing, and k_{eff} for the above condition when the storage is fully loaded and flooded with non-borated water. (9.1.2)

430.191

List the specific provisions included in the design of the spent fuel pool to comply with GDC 63, "Monitoring Fuel and Waste Storage" (e.g., pool liner leakage detection, water level monitoring and radiation monitoring systems). Identify the corrective actions on detection of loss of decay heat removal capability or excessive radiation levels. Note that for radiation monitoring systems, additionally referencing ABWR SSAR Subsections 11.5.2.1.2.1 and 11.5.2.1.3, if they are applicable, in ABWR SSAR Subsection 9.1.2.4 is sufficient. (9.1.2)

430.192

Provide the results and conclusions of the load drop analysis which considers dropping of one fuel assembly and its associated handling tool from a height at which it is normally handled above the spent fuel storage racks. ABWR SSAR Subsection 9.1.4.3 does not discuss compliance with GDCs 61 and 62; therefore, discuss the above compliance for the light load handling system. (9.1.4)

430.193

A "slack cable" signal (ABWR SSAR Section 9.1.4.3) is not considered sufficient indication of a fully seated assembly. Discuss whether positive vertical position indication will also be provided. (9.1.4)

430.194

ABWR SSAR Subsection 9.1.4.2.2.1, Reactor Building Crane, indicates that the crane can be used to move new fuel to the spent fuel pool and is also used to handle the spent fuel cask over the spent fuel pool and results of a failure modes and effects analysis demonstrating the adequacy of controls and interlocks to prevent compromising criticality or radiological safety. (9.1.4)

430.195

Clarify whether the system design includes interlocks (1) to ensure correct sequencing of the transfer operation in the automatic or manual mode, and (2) to prevent the refueling platform and the fuel handling platform moving in the transfer area during operations of the transfer system so that the transfer system will not be adversely affected by the presence of either platform. (9.1.4)

430.196

ABWR SSAR Tables 3.2-1 (page 3.2-28) and 9.1-2 differ in seismic classification identification for some fuel servicing equipment. Correct the discrepancy as appropriate. (9.1.4)

430.197

Provide an enlarged legible version of ABWR SSAR Figure 9.1-12, “Plant Refueling and Service Sequence”. (9.1.4)

430.198

ABWR SSAR Section 9.1.4 is confusing on the following details: (9.1.4)

(a) ABWR SSAR Subsection 9.1.4.2.3.7 and 9.1.4.2.3.8 refer to a fuel handling platform; but it is not described anywhere under that caption. It is not clear what constitutes the fuel handling platform and whether it is distinct from the refueling platform.

(b) ABWR SSAR Table 9.1-10 refers to three single-failure-proof cranes: the reactor building crane, refueling bridge crane and fuel handling jib crane. ABWR SSAR Subsections 9.1.4.2.7.1 and 9.1.4.3 refer to the automatic refueling machine (a gantry crane) and the spent fuel handling crane. It is not clear which of the above descriptors mean the same load handling device.

(c) Different subsections in ABWR Section 9.1.4 refer to the fuel storage pool, reactor building fuel storage pool, fuel pool and spent fuel pool. It is not clear whether all the above descriptors mean the spent fuel pool. Provide clarification on all the above. Also, provide layout drawings for all the storage pools, including the upper pool and the transfer canal.

430.199

Include the single-failure-proof characteristics of all cranes used in light load handling (note that ABWR SSAR Subsection 9.1.4.1 mentions only hoists on the refueling platform). (9.1.4)

430.200

ABWR SSAR Subsection 7.6.1 does not provide an evaluation of the radiation monitoring equipment for the refueling and service equipment as stated in ABWR SSAR Subsection 9.1.4.5.4. Provide the above information. If it is covered by some other radiation monitoring systems (e. g., area radiation monitoring system and/or process and effluent monitoring system

or both), include reference to those systems and the applicable SSAR Sections in SSAR subsection 9.1.4.5.4. (9.1.4)

430.201

The interface criteria of ABWR SSAR Section 9.2.15 does not include the required interface criteria for the design of the potable and sanitary water system. To meet the requirements of GDC 60, the design of this system should not allow for interconnections between the potable and sanitary water system and systems having the potential for containing radioactive materials. Protection should be provided through the use of air gaps, where necessary. Add these design criteria, as interfaces, under ABWR SSAR Section 9.2.15. (9.2.4)

430.202

Include the following interfaces besides what have been already specified for ensuring the ultimate heat sink (UHS) capability: (1) Design to accommodate single failures of passive components in electrical systems. (2) Protection of safety-related portions from adverse environmental conditions including those resulting from piping failures. (3) Time duration of UHS cooling capability availability. (9.2.5, 9.2.15)

430.203

The ultimate heat sink beat load requirements are identified by reference to ABWR SSAR Table 9.2-4. This set of three tables (9.2-4a, 9.2-4b and 9.2-4c) identifies heat loads for each of the three reactor building cooling water divisions. These tables do not consider the case of a reactor shutdown at 4 hours after a blowdown to the main condenser. Inclusion of the above may require a higher heat load dissipation capability for the UHS than what has been currently estimated (see GE’s response to Question No. 440.73). Revise the tables as appropriate considering the above case and provide the heat load requirements based on the revised tables for the ultimate heat sink (e. g., the sum of the heat loads for all three divisions, 2 of 3). Are there additional heat loads associated with the UHS not carried by the reactor building cooling water system? (9.2.5)

430.204

The requirements of 10CFR52 include the need for a conceptual design for systems not considered to be within the design scope of a standard nuclear power plant. No such conceptual design has been included as part of the ABWR SSAR for either the UIHS or the interfacing service water system. Provide conceptual designs for the UHS and the interfacing service water system. (9.2.5)

430.205

The make-up water preparation system is identified as outside the scope of ABWR standard plant. This system should meet the requirements of Position C.2 of Regulatory Guide 1.29. Provide an interface requirement that the failure of the make-up water preparation system will not result in the failure of any safety-related structure, system or component. (9.2.8)

430.206

Clarify how the turbine building cooling water (TCW) system meets Regulatory Guide 1.29, Position C.2 with respect to seismic requirements for non-safety-related systems that due to their failure during seismic events may adversely impact structures, systems or components important to safety. (9.2.14)

430.207

For the TCW system, provide information on the following items: (9.2.14)

(1) Effect of any system component failure including rupture of the atmospheric surge tank on structures, systems or components important to safety.

(2) Required total cooling water flow and available cooling water flow; total heat output by turbine building auxiliary equipment and available capacity of the TCW heat exchangers.

(3) Power cycle heat sink to which the heat from the TCW system is rejected.

430.208

The system diagrams lack sufficient detail to ascertain whether or not connections between the TCW system and safety-related water systems exist. Provide assurance that no such connections to safety-related-systems are provided or identify such connections and the isolation capabilities provided. Isolation capabilities should include the use of equipment that is at least Quality Group C and Seismic Category I. (9.2.14)

430.209

Only ABWR SSAR Sections 6.2-5 and 6.7 discuss the Atmospheric Control System (ACS) and High Pressure Nitrogen System (HPINS); therefore, correct SSAR Section 9.3.1 which refers to the wrong SSAR sections for discussion of the above systems. Also, provide information on the following items for the ACS:

(1) Clarification on applicability of system design criteria 9, 10, and 11 (protection against single active component failure, missiles, dynamic effects due to piping failures, tornado-missiles, flooding and seismic events) to all non-safety class system components (e.g. nitrogen storage tanks, vaporizers, applicable valves and piping, and instrumentation). (For these criteria, see SSAR Subsection 6.2.5.1). Specify, if some of the design bases for the ACS identified in Subsection 6.2.5.1 are applicable only for the safety-related components of the system, correct the subsection as appropriate.

(2) Justification for location of the inboard primary containment isolation valves outside the containment, which is a deviation from GDC 56 “Primary Containment Isolation. “The affected lines are (1) 2-inch N_2 makeup lines to the drywell and wetwell, (2) 22-inch purge suction lines to the drywell and wetwell (used for primary containment

inerting or de-inerting and connected to a common 16-inch N_2 supply line), and (3) 2-inch and 22-inch purge exhaust lines from the drywell and wetwell. We find your response to Question Nos. 430.35 and 430.42 does not include justification for deviation from GDC 56 requirements for the above lines nor deviations from GDC 56 or 55 “Reactor Coolant pressure Boundary Penetrating Containment” requirements for other applicable lines. Include justification for deviations from applicable GDC for other lines listed in ABWR SSAR Table 6.2-7.

(3) SSAR Subsection 6.2.5.2.7, which discusses the Flammability Control System (FCS), does not provide sufficient details for us to conclude that the system complies with the requirements of TMI Action Item II.E.4.1, “Dedicated Hydrogen Penetrations” of NUREG-0737. Therefore, include the system in Table 3.2-1 and provide details such as; how long after LOCA and at what concentration level of hydrogen the recombiner has to be activated; line sizes as related to flow requirements; and duration of recombiner operation. Also, identify interface requirements for referencing applicants with regard to the external recombiners (e.g. development of procedural provisions to assure availability of possibly shared portable hydrogen recombiners between sites on a timely basis and coordination of surveillance programs in accordance with SRP 6.2.5 acceptance criterion II.12).

(4) ABWR SSAR Tables 6.2-7 and 6.2-8 give a line size of 4 inches and 6 inches respectively for the FCS return line; Table 6.2-7 and Figure 6.2-40 show location of FCS primary containment inboard isolation valves inside the containment and outside the containment respectively; SSAR Sections 6.2.5.2.7 and 19.A.2.12 indicate portable and permanently installed recombiners, respectively. Resolve all the above inconsistencies. Also, of the location of all the primary containment isolation valves for the system is outside the containment, justify the deviation from the GDC 56 requirement for the system inboard isolation valves. (9.3.1)

430.210

Clarify which portions of the high pressure nitrogen gas supply system (nitrogen storage bottles, system piping including tie lines between safety-related divisions and non-safety-related division, valves, instrumentation and controls) are safety-related. (9.3.1, 6.7)

430.211

ABWR SSAR Figure 6.7-4 shows only one motor-operated isolation valve on each of the tie lines between each safety-related division and the common non-safety-related division of the high pressure nitrogen gas supply system (MO-F012A and B). The tie piping portion between the two isolation valves is presumably non-safety-related. Explain how essential nitrogen demand will be met during a situation when there is a pipe rupture is one safety-related division (initiating event), single active component failure in the other safety-related division (e. g., isolation valve on the applicable tie line is open) and a pipe break in the non-safety-related

portion of the tie lines (if there is such a portion). Alternately, provide two safety-related automatic isolation valves in series on each tie line. (9.3.1, 6.7)

430.212

Provide an FMEA for the Nitrogen Gas Supply System. (9.3.1, 6.7)

430.213

Include the nitrogen gas supply system in the ABWR System classification summary Table 3.2-1. (9.3.1, 6.7)

430.214

Contrary to what has been stated in ABWR SSAR Subsection 6.7.1, there is only one non-safety-related continuous nitrogen supply portion common to the two essential supply divisions (See Figure 6.7-1). Correct Subsection 6.7.1 as appropriate and discuss the effect of loss of nitrogen supply via the non-safety-related portion to all the equipment and components identified in SSAR Section 6.7.1 (e. g., Pneumatically operated valves and instruments inside the primary containment vessel) during normal operation. Clarify whether the pneumatic accumulator which provides the backup operating gas for the main steam isolation valve (See SSAR Subsection 5.4.5.2) is safety-grade for each valve. If not, justify the design. (9.3.1, 6.7)

430.215

Provide enlarged and legible piping and instrumentation diagram for instrument air and service air systems (SSAR Figures 9.3-6 and 9.3-7), which clearly indicate all the components served, safety and non-safety-related portions, and isolation provisions between the safety and non-safety-related portions; a table showing instrument air consumption during normal plant operation. Explain the statements in SSAR Subsections 9.3.6.1.1 and 9.3.7.1.1 which indicate that the containment penetrations (secondary containment penetrations) for the instrument air and service air systems are equipped with sufficient isolation valves to satisfy single failure criterion (the SSAR figures do not indicate this). Under the “Location” column for Item P.4 (Instrument/Service Air Systems), Sub-item 5 of ABWR SSAR Table 3.2-1(Page 3.2-33), include turbine building, radwaste building and service building since some of the components of these systems are located in these buildings. Also identify the design feature of safety-related air-operated valves outside the containment to handle the loss of air supply by the non-safety-related instrument air system during plant operation. (9.3.1, 9.3.6, 9.3.7)

430.216

Discuss the specific features provided (e.g. pre and after filters associated with compressors, particle size, dryer) for ensuring that air or nitrogen supplied by each of the applicable systems to components important to safety (e.g., MSIVs; SRVs; scram valves which are located outside the containment) meet the quality requirements (clean, dry and oil free) of ANSI MC 11.1-1976 standards. In this context, the staff finds GE’s justification for limiting particle size to 5 microns in the air stream at the instrument (the particle size is mentioned only for the instrument air system) instead of 3 microns as required by the above standards unsatisfactory (see Generic Letter 88-14 “Instrument Air Supply System Affected Safety-Related Equipment”). Note that

the staff will accept higher than 3 microns only if the larger size is supported by supplier's data for all the safety-related equipment or components that are supplied compressed air or nitrogen for their operation and there is assurance that the larger size will not cause any equipment or component degradation with aging. Also, discuss how all the above systems meet the guidelines of Regulatory Guide 1.68.3, "Preoperational testing of Instrument and Control Air Systems. "Include the atmospheric control system since it supplies nitrogen for safety-related components via the non-essential portion of the nitrogen gas supply system during normal power operation. Include the service air system since it supplies air to safety-related components inside containment during refueling. Identify applicable interface requirements for all the nitrogen or air supply systems with regard to fluid quality and preoperational testing requirements. (6.2.5, 6.7, 93.1)

430.217

Provide description and figures showing how the four compressed gas systems (atmospheric control, nitrogen gas supply, instrument air, and service air systems) are interconnected. Include isolation capabilities, if applicable, between the essential divisions of nitrogen gas supply system, and instrument air and service air systems.

430.218

Clarify whether the instrument air system supplies backup air to the nitrogen consumers located inside the primary containment during normal plant operation when the nitrogen gas supply pressure drops below the specified setpoint. If so, justify supply of backup air instead of backup nitrogen inside the containment during normal operation when containment has to be maintained inert. (9.3.1, 9.3.6)

430.219

Clarify whether both air compressors of the service air system operate simultaneously whenever the demand for service air exceeds 50% of the peak air consumption. (9.3.1, 9.3.7)

430.220

Compressed air or nitrogen supply systems designed to supply fluid to equipment or components located inside the containment for their operation at no more than design basis accident peak containment pressure will not be able to perform their intended function at higher containment pressures which may result under degraded core conditions. This, in turn, may compromise the operation of the subject components. Address the above concern as it relates to the design of compressed air and nitrogen gas systems. (6.7, 9.3.1)

430.221

Provide system P&ID for radioactive drain transfer system, which clearly show the safety-related portions of the system and the primary containment isolation valves. Provide a description of the loop seal design for the secondary containment penetrations for the system which includes (but is not limited to) survivability under various modes of reactor conditions (e.g. transients, accidents) and safety classification (seismic category and Quality group). Also, provide design and expected flow capacities and sump capacities. (9.3.3, 9.3.8)

430.222
Provide information regarding the effects of blockage in any portion of the drain system, including potential overflow paths. (9.3.3, 9.3.8)

430.223
Are the level switches for “each” sump of the radioactive drain transfer system (e.g. ECCS pump rooms, fuel handling area, steam tunnel) redundant and safety related. Do the level switches annunciate an alarm and provide level indication in the control room in case of rising water level? If they are not designed as stated above, justify the design. Also, include the sump level switches in ABWR SSAR Table 3.1-1 under “Radioactive Drain Transfer System”. Further, identify which flow transmitters located in the secondary containment under “Leak Detection and Isolation System” in SSAR Table 3.2-1 are non-safety-related. (9.3.3, 9.3.8)

430.224
ABWR SSAR Subsection 9.3.8.2.1 indicates that the capacity of the nonsafety-related radioactive drain transfer system, in conjunction with the placement of safety-related equipment on raised pads or grating, precludes the adverse consequences of flooding on safety-related equipment and components. However, SSAR Subsection 3.4.1.1.2 states that the ABWR design does not take any credit for operation of the drain sumps to provide flood protection. Resolve the above inconsistency, realizing that the drain transfer system has to be safety-related is its operation is to be credited for flood protection of safety-related equipment and components. (9.3.3, 9.3.8)

430.225
Identify the system design features and their safety classification (i.e. seismic category, quality group) provided to prevent backflooding of safety-related equipment rooms (e.g. ECCS equipment rooms). (9.3.3, 9.3.8)

430.226
Provide an interface requirement for the drainage systems for non-radioactive liquid waste prohibiting any conditions to the radioactive drain transfer system. (9.3.3, 9.3.8)

430.236
Since the service building is a nonsafety-related structure, justify its inclusion in the list of locations of some electrical modules and cables performing a safety-related function and some safety-related valves and dampers of the HVAC systems. Also, justify nonsafety quality group classification for “other safety-related valves and dampers” for HVAC systems (see ABWR SSAR Table 3.2-1 page 3.2-29).(9.4)

430.237
Explain the words “high efficient section” occurring in SSAR Subsection 9.4.1.1.3, second paragraph. If the above words mean HEPA filter, include it in SSAR Figure 9.4-1, and provide a table listing compliance status including justification for non-

compliance with each of the applicable guidelines identified in Positions C.1 and C.2 of Regulatory Guide 1.140 for control building normal ventilation exhausts. (9.4)

430.238

Clarify whether (1) the two redundant safety-related trains of the control room equipment HVAC system are totally independent and whether each has 100% capacity and (2) the three subsystems of the essential electrical HVAC system (SSAR Subsection 9.4.1.2.3) are totally independent so that failure of any one subsystem will not compromise the availability of the remaining two subsystems. Also, explain what Essential Chiller Room C (SSAR Subsection 9.4.1.2.3) means since the HECW system presumably has only two safety-related chiller trains. (9.4.1)

430.239

Provide complete system P&IDs including safety classification changes (i.e., seismic category and quality group) for the control building HVAC system (i.e., SSAR Sections 9.4.1.1 and 9.4.1.2). The P&IDs should show among other things (1) monitors located in the system intakes that are cable of detecting radiation and smoke, (2) capability for isolation of nonessential portions by two automatically actuated dampers in series and (3) provisions for isolation of the control room upon smoke detection at the air intakes. Also, provide complete flow diagrams for all modes of control building HVAC system operation (i.e., normal, accident, smoke/toxic gas removal) shoving among other things flow rates and component description tables for the building HVAC system (SSAR Figure 9.4-1 is illegible in parts and is also incomplete). (9.4.1)

430.240

SSAR Subsection 9.4.1.1.3 states that the emergency recirculation system includes an electric heating coil whereas SSAR Figure 9.4-1 shows only a hot water system connection to a heating coil. The above figure additionally shows three HECW divisions whereas SSAR Subsection 9.2.13 mentions only two HECW divisions. Resolve the above inconsistencies. Also, clarify whether the normal recirculation unit and the hot water system are safety-related, since their availability during the emergency mode of operation is vital to maintaining proper environmental conditions in the control room and at the safety-grade filter train. (Note that there is no description of the hot water system in the SSAR. This should be provided.) (9.4.1)

430.241

Clarify whether the system air intakes are provided with tornado missile barriers. (9.4.1)

430.242

For the turbine building ventilation system, provide (1) complete system P&ID including safety classification changes and isolation and monitoring devices, (2) complete system flow diagrams showing description tables. Also, identify the corrective operator action following annunciation of alarms upon detection of high radiation in the building ventilation exhaust. (9.4.4)

430.243
For the reactor building ventilation system, provide the following:

430.243a
Complete system P&IDs including safety classification changes, isolation and monitoring devices for secondary containment (e.g., radiation monitors in the secondary containment ventilation exhaust, spent fuel pool and essential equipment room area exhausts), essential electrical equipment, essential diesel generator, drywell purge and reactor internal pump control panel room HVAC subsystems.

430.243b
Some of the SSAR figures (e.g., Figures 9.4-3, 9.4-4) have illegible portions; there is no figure in the SSAR for the mainsteam/feedwater tunnel HVAC subsystem; SSAR Figure 9.4-3 for secondary containment HVAC subsystem does not show servicing of rooms housing redundant equipment for some essential systems; and the figures do not specify flow rates. Provide enlarged and legible size complete flow diagrams showing flow rates among other things for each subsystem (for guidance in contents for requested response, see GESSAR-II HVAC system flow diagrams provided in the GESSAR-II SAR).

430.243c
Component description tables for each subsystem.

430.243d
FMEA for each subsystem.

430.243e
Description of isolation devices including safety classification, redundancy and source of power to the devices for all nonsafety-related HVAC subsystems that interface with safety-related structures, systems and components (SSC) (e.g., secondary containment HVAC subsystem, drywell purge supply/exhaust subsystem).

430.243f
Specific design characteristics for meeting GDC 4 requirements for safety-related HVAC subsystems.

430.243g
Table listing compliance status with each of the applicable guidelines of Regulatory Guide 1.140, Positions C.1 and C.2 including justification for non-compliance for the normal ventilation exhausts from the secondary containment and drywell purge subsystems (SSAR Subsection 9.4.5.1.2 refers to filters in the secondary containment normal exhaust system, but does not discuss what kind these are).

430.243h

Discussion of smoke removed operation for applicable HVAC subsystems including how the affected area will be isolated from other unaffected plant areas. Also, include the impact of applicable HVAC subsystems in safe or alternate shutdown capability for a fire event in a plant area serviced by one of the applicable subsystems.

430.244

ABWR Subsection 9.4.5.4.2 states that each divisional HVAC system consists of two power supply fans, two exhaust fans, and two recirculation units. However, SSAR Figure 9.4-4 shows only one recirculation unit per division. Also, the figure shows three HECW divisions supplying chilled water to the respective division room coolers; but SSAR Section 9.2.13 describes only two divisions for the HECW system. Resolve the above discrepancies realizing that the safety-related support systems for three diesel generators have to be completely independent of each other. (9.4-5)

430.245

Confirm that each supply and exhaust fan (of the essential electric equipment room HVAC System) mentioned above is a 100% capacity fan. (9.4.5)

430.246

Discuss how the essential electric equipment HVAC subsystems meets GDC 17 “Electric Power Systems” as it relates to the protection of essential electrical components of the subsystem from failure due to the accumulation of dust and particulate materials (see SRP Section 9.4.5, Acceptance Criterion II.4 for required contents of response to this item). (9.4.5)

430.247

Subsection 9.4.5.4.5 does not discuss temperature contro1. Provide a discussion of the method and instrumentation provisions for temperature control. (9.4.5)

430.248

Provide a discussion of the means used for maintaining the rooms cooled by the essential electrical equipment HVAC system at positive pressure. (9.4.5)

430.249

Provide assurance that the air intake elevation for the essential diesel generator HVAC system is greater than 20 feet above grade or discuss the methods for protecting electrical panels from dust and particulate materials. (9.4.5)

430.250

ABWR SSAR Subsection 9.4.5.5.2 states that the two supply fans for each of the three diesel generators take air from the outside and distribute it to the diesel generators. Clarify whether there is a common header for all the diesel generators for intake air. If there is, justify such a design. (9.4.5)

430.251
Provide drawings for the drywell purge supply/exhaust system and a discussion of the interfaces to the secondary containment HVAC system and to the standby gas treatment system. (9.4.5)

430.252
Discuss the sensor location and actuation setpoint for the exhaust radiation monitor for the drywell supply/exhaust system as they relate to preventing unanticipated radioactive releases. (9.4.5)

430.253
Since there are Separate wetwell purge supply/exhaust system for the ABWR, include a description of that system in the SSAR. Note that all the information requested above for the drywell purge system should be included in the description of the wetwell purge system. (9.4.5)

430.254
ABWR SSAR Subsection 9.4.5.6.1.2 states that the drywell purge system only operates during plant shutdown. Correct the above statement since it will operate also during inerting, deinerting or pressure control of the primary containment. Also, discuss how both the drywell and wetwell purge supply/exhaust subsystems together meet Branch Technical Position CSB 6-4 “Containment Purging During Normal Plant Operation.” (9.4.5)

430.255
ABWR SSAR Subsection 9.4.5.1.2 states that two fan coil units provide cooling to the steam tunnel. Explain how the air is cooled. (9.4.5)

430.256
ABWR SSAR Subsection 9.4.5.8.2 states that each division of the reactor internal pump (RIP) control panel room HVAC subsystem contains two recirculation units. This does not agree with Figure 9.4-5. Resolve this discrepancy. (9.4.5)

430-257
ABWR SSAR Subsection 9.4.5.8.3 addresses the non-essential equipment HVAC system instead of the RIP control panel HVAC system. Provide a safety analysis which addresses the proper system, including a discussion of the effects of loss of ventilation on the RIP control panel. (9.4.5)

430.258
For the radwaste control room and balance of the radwaste building HVAC systems, provide (1) complete P&IDs showing safety classification changes, isolation and monitoring devices, (2) complete flow diagrams showing among other things flow rates, and (3) component description tables. Also clarify whether any affected space is isolated by safety-related devices. (9.4.5)

430.259
ABWR SSAR Subsection 9.4.6.2.2 states that one radwaste building HVAC supply and exhaust fan are normally operating and the other of each type (i.e., for the radwaste control room and the balance of the radwaste building) is on standby. SSAR Subsection 9.4.6.3 mentions provisions for automatic start of the standby unit. However, SSAR Subsection 9.4.6.5.2 indicates that only an alarm is actuated by low flow in the exhaust fan discharge duct, and that ventilation must be restarted manually. Clarify whether the standby fan is started on failure of the operating fan. If not, provide justification. (9.4.6)

430.260
Provide a failure modes and effects analysis for the radwaste building HVAC system which shows that the normal direction of air flow from areas of low potential contamination to areas of higher contamination will not be reversed for the failure of any active component. (9.4.6)

430.261
For both of the radwaste building HVAC system zone exhausts, provide tables listing compliance status including justification for non-compliance with each of the applicable guidelines identified in Positions C.1 and C.2 of Regulatory Guide 1.140. (9.4.6)

430.262
For the service building ventilation system, provide complete system P&IDs including safety classification changes, isolation and monitoring devices, (2) component descriptions tables, and (3) compliance with applicable guidelines of Regulatory Guide 1.140 for the system exhaust. Also, provide legible and enlarged portions of the SSAR Figure 9.4-7 which are currently illegible; include flow rates in the figure. (see Figure 9.4.8)

430.263
Provide enlarged and legible versions of the drywell cooling system P&ID (SSAR Figure 9.4-8).

430.264
Identify the HVAC system that will service the remote shutdown panel area that will be used for providing alternate shutdown capability following certain fire events. (9.4)

430.265
Identify interface requirements as they relate to HVAC systems for plant areas which do not fall within the ABWR design scope but which may impact the SSC that are within the ABWR scope. Also, provide interface requirements for the technical support center (TSC) HVAC system. (9.4)

430.266
Provide system layout diagrams for the diesel generator support systems. These diagrams should be of sufficient detail that component location can be determined and the accessibility of equipment for test and maintenance can be evaluated. Physical separation between individual

subsystems of each support system serving the three diesel generators should be explicitly stated in the respective SSAR section. (9.5.4–9.5.8)

430.267

For each diesel generator support system, provide sufficient information on how each system is protected against the effects of failure of any high or moderate energy piping located near the system components.

430.268

Provide a failure modes and effects analysis including loss of offsite power situation for the components of each diesel generator support system.

430.269

Provide information on location and mounting of controls and instrumentation for all the diesel generator support systems in so far as they relate to protecting the system components from adverse effects due to engine vibration during engine operation (see NUREG/CR-0660 "Enhancement of on-site emergency diesel generator reliability," Recommendation C.6).

430.270

ABWR SSAR Table 3.2-1 shows that some safety-related components of the diesel generator support items are located outdoors onsite (see SSAR page 3.2-24.1, Items R3.3, 4 and 5). Explain how these are protected against the effects of SSE, flood and tornado-missile.

430.271

The staff agrees with GE that the keep-warm heaters and associated pumps of the diesel generator lubrication system and the air compressors and motors of the diesel generator starting air system need not be nuclear safety class. Except for the above, the staff requires that all piping and components of all the support systems up to the engine interface should be designed, fabricated and installed in accordance with ASME code, Section 111, Class 3 requirements. The staff considers the engine interface as being the first connection off the engine block—flanged, welded or screwed. Clarify whether the design of the support systems meets the above requirement. If it does not, provide justification for the deviations. Also, explain how the safety-related portions of the support systems are protected from the effects of failure of non-safety-related portions of the systems.

430.272

Provide P&IDs for the diesel generator fuel oil and transfer system that include safety classification changes and level, temperature and pressure sensors among other things. (9.5.4)

430.273

Discuss the Provisions for measuring fuel oil temperature and pressure and maintaining it within recommended limits.

430.274

Provide information on the following in the system description:

(1) Type of transfer pump.

(2) Design features for protecting diesel generator fuel oil fill and vent lines from the effects of SSE, flood and tornado-missile.

(3) Clarification on provision of a stick gauge connection for each tank.

(4) Internal and external corrosion protection features for exposed and buried portions of the system including the storage tanks (see Regulatory Guide 1.137 “Fuel Oil Systems for Standby Diesel Generators,” Position C.l.g).

(5) Provisions for removal of accumulated water from the fuel storage tanks (see NUREG/CR-0660 Page V-16, Recommendation “a”).

(6) Precautions after fill-up of oil tank to minimize potential causes and consequences of fires and explosions (see SRP Section 9.5.4, Item III.7).

430.275

An event may occur requiring the replenishment of fuel oil in the storage tank without interrupting the operation of the diesel generators. This, in turn, may result in turbulence of the accumulated sediment (at the bottom of the storage tank). Further, the duplex filters in the transfer pump discharge piping may not be able to handle the above problem. Therefore, describe additional features that will be provided for preventing turbulence of accumulated sediment during filling of the storage tank, so that uninterrupted supply of fuel oil will not be compromised (see Regulatory Guide 1.137, Position C.2.g).

430.276

Discuss system compliance (including justification for noncompliance), if applicable, with Positions C.l.e, f and C.2.a, b, d, e, f and h of Regulatory Guide 1.137 (the staff notes that the above guidelines are not addressed in SSAR Section 9.5.4).

430.277

Identify the power source for the jacket water circulating system.

430.278

Provide P&IDs for the diesel generator cooling water system which include safety classification changes. (9.5.5)

430.279

Provide a table of design flow and heat removal requirements for the diesel generator cooling water system. Also, provide the design heat removal capacities of all the coolers or heat exchangers in the system. SSAR Figure 9.2-le shows intercoolers, lube oil coolers and filtered

water coolers; SSAR Section 9.5.5.2 however, uses different terminology to identify some of the above (e.g., air intercooler, jacket water heat exchanger). Identify clearly all the heat exchangers or coolers in the system and clarify whether the combustion air is also cooled by the system. (9.5.5)

430.280

(9.5.5) Provide information on the following:

(1) Type of jacket water circulating pumps.

(2) Clarification as to whether the system includes a motor-driven jacket water keep-warm pump; describe the keep-warm feature of the system.

(3) Identification of all system heat exchangers or coolers where heat is rejected to the RBCW system (SSAR Subsection 9.5.5.2 states “jacketed manifold and a heat exchanger which is furnished with RCW” whereas SSAR Figure 9.2-1e shows supply of RCW to two intercoolers, one lube oil cooler and one filtered water cooler for each diesel generator).

(4) Type of temperature sensors (“Amot” brand or equal with an expanding wax type temperature sensitive element?—see NUREG/CR-0660, Page V-17, Recommendation under Item 4).

(5) Clarification as to whether the system can be vented to assure that all spaces in the closed loop are filled with water (see SRP Section 9.5.5, Item III.2).

(6) Function of the filtered water cooler shown in SSAR Figure 9.2-1e.

(7) Provisions for isolating non-safety-related portions from safety-related portions of the system.

430.281

SSAR Subsection 9.5.5.4 gives little information regarding periodic inspection (e.g., accessibility of areas) and testing (e.g., structural and leak tight integrity of the components, active components and system as a whole). Discuss how the system complies with GDCs 45 and 46 regarding inspection and testing of the system (Note that layout diagrams alone are not sufficient and that these should be supplemented by descriptive information).

430.282

Are the diesel generators capable of operating at design loads without secondary cooling (i.e., by the RBCW system) in excess of time needed to restore RBCW supply to the diesel generators cooling water system following a loss of offsite power?

430.283
Demonstrate by analysis that normal system coolant leakage over a 7-day period will not exceed the excess amount of coolant contained in the system expansion tank and/or cause loss of positive suction bead to jacket water circulating pumps.

430.284
Identify the operating procedures to ensure that the diesel generators and the associated cooling water system can perform their design functions for extended periods when less than full electrical power generation is required without degradation of their performance or reliability (note that operating procedures identified in SRP Section 9.5.5, Item III.7 are acceptable).

430.285
Discuss the provisions for controlling the dew point of diesel generator starting air; also, identify the temperature to which the dew point would be controlled.

430.286
Provide P&IDs for the diesel generator starting air system that include safety classification changes, pressure gauges, relief valves, drain valves and isolation valves among other things. Provide system design requirements such as compressor capacity, power source, and receiver tank capacity. Also, identify the design features which will maintain the receiver pressure within an allowable range. (9.5.6)

430.287
Include devices to crank the engine as recommended by the engine manufacturer as one of the components of the system. Also, identify the air start requirements with regard to the duration of the cranking cycle and number of engine revolutions (see SRP Section 9.5.6, acceptance criterion III.g).

430.288
Identify system provisions for periodic or automatic blowdown of accumulated moisture and foreign material in the air receivers and other critical points of the system. In this context, the staff notes that NUREG/CR-0660 has identified water in the starting air as the “root cause” for most of the troubles reported for the system and has, therefore, strongly recommended refrigerated starting air driers with a minimum flow rate of 250 SCFM between the compressors and the receivers and automatic water drains (see Page V-4 of the NUREG). Clarify whether the system design includes the above features for water removal and if not, provide justification.

430.289
Discuss how the diesel generator support systems meet the NUREG/CR-0660 recommendations 2.a, 2.b, and 2.d (Page V-6) and 5 (Page V-18) on concrete floors painting) with regard to protection of these systems from the adverse effects of dust and dirt.

430.290

Clarify whether the fill connection for the lube oil supply tank is located in the locked diesel generator room or discuss the provisions for preventing lube oil contamination via the fill pipe.

430.291

Provide P&IDs for the diesel generator lubrication system that include safety classification changes, temperature, pressure and level sensors among other things. Include the pre-lube provisions in the P&IDs. (9.5.7)

430.292

Provide design criteria (pump flows, operating pressure, temperature differentials, cooling system heat removal capabilities, electric heater characteristics) for the diesel generator lubrication system.

430.293

Describe how the diesel generator lubrication system (1) complies with NUREG/CR-0660, recommendations 1 and 2 (see Pages V-9 and V-10 of the NUREG) regarding duration of the pre-lube period and starting and stopping of the pre-lube pump, and (2) precludes excessive pre-lubrication of the diesel engine turbocharger. In this context, clarify whether the keep-warm oil circulating pump can perform the function of the pre-lube pump. Also, provide information on how the pre-lube operation will be monitored. Note that the staff requires the monitoring/alarm circuit for the pre-lube system to be safety-related (Class 1E) to provide reliable indication of the system operation so that the operator can perform timely corrective action in case of failure.

430.294

Describe system protective features to prevent unacceptable crank case explosions and to mitigate the consequences of such an event.

430.295

Identify all the components in the flow paths for the diesel generator combustion air intake and exhaust system (e.g., air intake silencer, turbocharger, combustion air aftercooler). If the above include flow control devices (louvers, dampers), can the system function if there is failure of a single active component?

430.296

NUREG/CR-0660 recommends (Page V-15, recommendations l.a and l.b) that the piping for the diesel generator room ventilation air should be separate from that used for the combustion air and that the supply of the combustion air should preferably be through piping directly from outside the building and at least 20 feet from the ground level through proper filters. ABWR SSAR Subsection 9.5.8.2.1 states that each diesel engine takes combustion air from its own diesel generator room which, in turn, receives its air supply from the room air ventilation system. Explain why an advanced design like ABWR does not comply with the NUREG recommendations in so far as they relate to keeping the ventilation and combustion air supplies separate. Also, explain how the air exhaust silencers mounted at the roof of the reactor building

are protected from tornado missiles. Further, contrary to what has been stated, reactor building arrangement drawings in SSAR Section 1.2 do not show intake and exhaust locations for the system; include these locations in the applicable drawings.

430.297

Discuss the ability of the intake filters to provide sufficient filtered combustion air for the entire time period when emergency power is required assuming maximum particulate concentration at the intake.

430.298

Provide a system P&ID showing safety classification changes for the diesel generator combustion air intake and exhaust system. (9.5.8)

430.299

Identify the diesel engine operation procedures that will minimize or avoid incomplete combustion (see NUREG/CR-0660, recommendation B3a, Page V-11). (9.5.8.)

430.315

ABWR SSAR Section 9.5.1 provides fire hazards analyses for fire areas the reactor building only. Additionally, the section does not describe the specific reactor shutdown method that will be used for each of the fire areas in the reactor building [see Response A2]. (9.5.1) Provide the following:

(1) Fire hazards analysis for each fire area outside the reactor building which is within the scope of the standard design (e.g., control building, turbine building, radwaste building, service building) [see Response Al]. Note that an receipt of such information, additional information may be requested.

(2) Description of specific reactor shutdown method for each fire area. The description should discuss how the chosen method provides safe shutdown or dedicated shutdown, whichever is applicable for the given area, that is, how the shutdown method meets Positions C.5.b or C5.c of BTP CMEB 9.5-1 “Guidelines for Fire Protection for Nuclear Power Plants” of SRP Section 9.5.1 “Fire Protection Program” [see Response B1]. Specifically, the response for each fire area should include among other things, an associated circuit analysis (that is, how the common bus, common enclosure and spurious signal concerns including high/low pressure interface breaches will be eliminated) [see Response B2], available shutdown equipment including cables [see Response B3], required operator actions and the time when these have to be completed [see Response B4], and required repairs [see Response B5], if any, for achieving cold shutdown within the allowed time. Additionally, for the control room, the response should identify the specific design provisions to ensure the capability to transfer control of needed hot shutdown equipment to a remote shutdown panel without recourse to any hot shutdown repair [see Response B6].

(3) Lighting and communications provisions as they relate to the fire protection program for the ABWR. Your response should indicate how the program meets the specific guidelines stated under Positions C.5.g(l) through g(4) of BTP CMEB 9.5-1 [see Response C]. Note that cross referencing ABWR SSAR Sections 9.5.2 ”Communication Systems” and 9.5.3 “Lighting and Servicing Power Supply System” will not be an adequate response, since these sections do not discuss all the above guidelines.

(4) Interface requirements for referencing applicants for fire areas not within the scope of the standard design (e.g., ultimate heat sink area). For such areas, the interface requirements should call out for applicable information requested in Items (1), (2), and (3) above [see Response D].

430.316
The fire hazard analysis provided as Appendix 9A listed several components within the rooms of each fire area in the reactor building. However, specific cables (power and instrumentation) were not identified in the equipment listings (Tables 9A.b-1 and 9A.b.2). The failure of these cables will have to be included in a safe shutdown analysis. Additionally, the equipment listed in these two tables showed that equipment powered by separate divisions of AC power (Division 1 and 2 for example) are located in the same reactor building fire zones. From the information in Appendix 9A, it is not possible to determine if the failure of this equipment could affect the operability of required safe shutdown equipment in other fire areas. This equipment should be addressed in the safe shutdown analysis, including an associated circuit analysis. (9.5.1)

430.317
Section 9.5.1.2.1 should be expanded to include the fire protection water supply system.

430.318
Section 9.5.1.2.2 states that a manually operated carbon dioxide (CO_2) fire suppression system will be provided for the diesel generated rooms, including the day tank rooms. This does not correspond to the guidance provided in NUREG-0800, CMEB BTP 9.5-1, Section C.7.i which specifies automatic fire suppression for the emergency diesel generators. This section should be changed to show automatic fire suppression or expanded to justify how manual suppression provides either equivalent or superior protection.

430.319
Section 9.5.1.2.7 indicates a 30 second time delay discharge will be provided for the carbon dioxide fire suppression systems. This feature is appropriate for automatic systems but not for manually operated systems. This section should be changed to state that a time delay discharge will not be provided for manually operated systems, such as those systems provided for the emergency diesel generator rooms in case the manual systems are justified and retained in the ABWR design.

430.320

Section 9.5.1.2.8, smoke control refers to Section 6.4 and Subsection 9.4.5. (6.4)(9.4.5)(9.5.1)

(1) Section 6.4.4.2 states that “In the smoke removal mode, the purge flow through the control building provides three air changes per hour in order to sweep atmospheric contaminants out of the air.” An air change every 20 minutes will be effective for smoke control only for the very smallest fires in very large volumes. This section should be changed and expanded to describe how smoke will be removed from fire areas and to provide the technical bases, including test data, to support assumptions used in the smoke removal systems design.

(2) Section 9.4.5.xx states for the various areas where applicable, that “fire protection has been evaluated and is described in Subsection 9.5.1.” No descriptive material is contained in these subsections pertaining to smoke removal capability of the normal HVAC system.

430.321

Section 9.5.3.1.1(5)(f) should be expanded as follows: “Battery power supplies for lights in harsh environments (including high/low temperature areas) shall be located where the environment will nor degrade the batteries, or the batteries shall be qualified by test for the environment.”

430.322

Section 9A.2.1.1 should be expanded to include NFPA 20, “Centrifugal Fire Pumps.”

430.323

Section 9A ... 2.4(3) implies that walls with fire (9.5.1) resistance ratings less than 3 hours will be allowed as fire walls. This is not acceptable. Section 9A.2.4(3) should be clarified to clearly state that all fire barriers will have a minimum fire resistance rating of 3 hours.

430.324

Section 9A.2.4(11) states that redundant safe shutdown cables “are not permitted together in the same cable tray.” Actual separation of redundant safe shutdown cables should be specified, since literal compliance with this prohibition against the case of a single cable tray for redundant cables, could still result in an unacceptable condition.

430.325

Section 9A.3.1(8) states that one of the methods of protection for safety-related equipment and associated cabling is spatial separation (isolation). The staff does not recognize as acceptable for use in an advanced reactor design any method of protection which relies only upon spatial separation. We recognize the need for open communication between compartments inside containment in order to be able to relieve and equalize pressure following a high energy line break. Therefore, the use of structural walls inside containment as fire barriers to separate safety-related systems (cabling, components and equipment), even though such walls may not

fully enclose the equipment requiring separation, is acceptable in intent. Care must be taken in actual system layout, however, to assure that line-of-sight exposure between components requiring separation does not exist, and that a sufficient labyrinth is provided between the separated components to assure that fire spread does not occur. (9.5.1)

430.326

Section 9A.4.1.1.x under (9) “Consequences of Fire” for several different rooms or areas, states ‘smoke from a fire would be removed by the normal HVAC system, if it has not been isolated. If the normal HVAC system has been isolated, smoke removal is by the SGTS system.” In question 430.320 above we pointed out the limitations of ventilation systems that provided only three air changes per hour to function as a smoke removal system during fires. Other technical considerations aside (such as possibility of soot fouling), the capacity of the SGTS is so small relative to the building volumes served that it is clearly not capable of performing as an effective smoke removal system. Please describe how smoke removal from these areas will be accomplished if the normal HVAC system is not available. (9.5.1)

430.327

Section 9A.4.1.1.18 under (2) states no core cooling is provided in this room, while under (9) GE states that “the provisions for core cooling systems backup are defined in Section 9A.2.5.” Which statement is correct?

430.328

Section 9A.4.1.1.26 under (2) states yes—safety-related, however, under (9) states, “the function is not safety-related and its loss is acceptable.” Which statement is correct?

430.329

Section 9A.4.1.1.33, Subsection (9) “Consequences of Fire” states that “access to the operating handles for the manually actuated valves in the adjacent room would be temporarily reduced.” What does this mean and what is the postulated effect? Are manual actions in the adjacent room contemplated for safe shutdown?

430.330

Section 9A.4.1.2.10.Subsections (2) and (9) are mutually ambiguous. Is the equipment in this area safety-related and does it provide core cooling?

430.331

Sections 9A.4.1.2-18, 9A.4.1.2.19 and 9A.4.1.2.20 all state in Subsection (2) that equipment in each room is safety-related and provides core cooling. However, in each case, Subsection (9) states that, “the postulated fire assumes the loss of the function. The function is not safety-related and its temporary loss is acceptable.” Since these statements appear to be contradictory, please clarify.

430.332

Section 9A.4.1.2.25 states in Subsection (2) that equipment in the room is safety-related. However, Subsection (9) states that, “the function is not safety-related, therefore, the loss of the function is acceptable.” Which statement is correct?

430.333

Section 9A.4.1.2.32 has the same statements as Section 9A.4.1.2.25 (Question 430.332 above and the same question applies.

430.334

Section 9A.4.1.4.4 Diesel Generator A Room
Section 9A.4.1.4.10 Diesel Generator C Room
Section 9A.4.1.4.15 Diesel Generator B Room (9.5.1)

(1) See comments about CO_2 protection provided for Diesel Generator Room in Question 430.318 above relative to the description of the manual total flooding CO_2 system in Subsections (7).

(2) See Question 430.326 above relative to use of normal HVAC or SGTS systems for smoke removal. In addition to those concerns, we question the availability of the SGTS to the Diesel Generator Rooms.

430.335

Section 9A.4.1.4.26 states in Subsection (9) that the safety-related valves in this area will fail closed upon loss of actuation power. How is this accomplished? (9.5.1)

430.336

Section 9A.4.1.5.04 DG Control Panel and Service Corridor A (Room 514). Section 9A.4.1.5.08 DG Control Panel C and Service Corridor C (Room 532). Section 9A.4.1.5.16 DG Control Panel B and Service Corridor B (Room 522).

Subsection (9) for all three of these rooms states that fire could result in temporary loss of access to the B diesel generator HVAC room and that functional backup is provided by the A and C diesel generators. Although this subsection also states that continuous access to the HVAC is not required, we question how all three of these rooms (each related to a separate diesel generator unit) can cause loss of access to the same Train B diesel generator HVAC room.

430.337

Section 9A.4.1.5.23—Why is loss of the stack monitors as stated in Subsection (9) acceptable?

430.338

Sections 9A.4.1.6.02 through Section 9A.4.1.6.37—Question No. 430.326 above finds unacceptable reliance upon the SGTS for smoke removal when the normal HVAC system is not available. Subsection (9) in each of there sections references reliance upon the SGTS which we

understand to be the same as the SGTS. Therefore, the same concerns noted in Question No. 430.326 apply here also. (9.5.1)

430.339

Section 9A.4.1.7.01 states in Subsection (2) that systems in the room are safety-related but in Subsection (9) states that the functions are not safety-related. Which statement is correct? (9.5.1)

430.340

Section 9A.4.1.7.12 states in Subsection (9) that four divisions of the stack radiation monitors are located at the base of the stack and could be lost. Why is it acceptable to loose all four of these monitors?

430.341

Section 9A.5.1 reads as follows:

"9A.5.1 Piping Penetrations, Reactor Building Piping penetrations through the drywell shell have unique design considerations. The stress and containment requirements along with the temperature inputs to the concrete walls leave little design latitude. Experience has shown that some of these penetrations for high energy piping may not contain a 3-hour fire-resistive barrier such as have provided throughout the other ABWR buildings. Penetration details are not available at this stage of the plant design."

The staff understands this to mean that GE is proposing that we approve in advance deviations from the requirement to provide 3-hour fire rated penetration seals for certain as yet unidentified high energy piping. This is not acceptable. The applicant should state their intention to provide 3-hour fire rated penetration seals for all high energy piping or, as a minimum, state those conditions when such seals cannot be provided and what will be installed as a substitute. Sufficient technical detail must be provided to allow the staff to approve such deviations in principle. (9.5.1)

430.342

Section 9A.5.5.1 states that conduit from the separate divisions are separated from each other to meet IEEE 384. The IEEE 384 separation distances are primarily to prevent electrical signal interference between or among conductors. They do not necessarily provide adequate separation to satisfy fire protection needs. This statement should be expanded to assure that fire protection separation requirements are satisfied. (9.5.1)

430.343

Discuss conformance with requirements of 10CFR50.63 "Loss of All Alternating Current Power," as related to the support systems regarding (1) sufficient amount of water (condensate storage system), (2) sufficient flow path and delivery system (reactor core isolation cooling system). (3) decay heat removal capability (automatic depressurization system), (4) sufficient valve position indication and closure capability for containment isolation (containment

isolation system), (5) sufficient compressed air capacity for station blackout (SBO) components for core cooling and/or containment isolation (compressed air system), (6) suitable environmental conditions inside control room and other areas served to protect personnel and SBO equipment including instrumentation and controls (control room area ventilation system and engineered safety feature ventilation system), (7) common mode failures of sufficient fuel oil supply, transport and storage recharging capability, sufficient cooling and compressed air, adequate lubrication and air intake and exhaust for the diesel engine (emergency diesel engine support systems), if used as an alternate AC power source, and (8) battery capacity to assure that core is cooled and an appropriate containment integrity is maintained independent of preferred and onsite emergency AC power in the event of a station blackout for the specified duration and recovery therefore in accordance with the guidance of Regulatory Guide 1.155," station blackout, NUMARC 87-00 and NUMARC 87-00 supplementary guidance dated December 27, 1989.

440.101

SSAR Table 9.3-1 is not complete. Include pump flow and other parameters for all modes of operation. The existing Table 9.3-1 gives only test modes. (9.3.5)

440.102

In current BWR's, explosive valves are used at SLCS pump discharge. Why are they deleted? How is boron leakage into the reactor vessel prevented during testing? (9.3.5)

440.103

The ATWS rule states that "Each Boiling Water Reactor must have a Standby Liquid Control System (SLCS) with a minimum flow capacity and boron content equivalent in control capacity to 86 gpm of 13 weight percent-sodium pentaborate solution." (251 vessel, Ref: NEDE 31096-P-A) How does the ABWR design with 278 diameter vessel meet the requirements of the ATWS rule, 10CFR50.62? (9.3.5)

440.104

In the ABWR design, SLCS pump is started manually. But the ATWS rule 10CFR50.62 states that "The SLCS initiation must be automatic and must be designed to perform its function in a reliable manner for plants granted construction permits after July 26, 1984." How does the ABWR design satisfies the ATWS rule? (9.3.5.)

440.105

We understand that boron mixing tests were performed for optimizing the location of boron injection. Describe the test criteria and the test results. (9.3.5)

440.106

In SSAR Section 9.3.5.3, under criterion 26, it is stated that "The requirements of this criterion do not apply within the SLCS itself." Elaborate on this assumption. (9.3.5)

440.107

In SSAR Section 9.3.5.3, under criterion 27, it is stated that “this criterion applies no specific requirements onto the SLCS and therefore is not applicable.” Describe in detail the justification for the above statement. (9.3.5)

20.2.10 Chapter 10 Questions

281.15

In a letter from Thomas E. Murley, NRR, to Ricardo Artigas, G.E. dated August 7, 1987, the staff provided the ABWR licensing review bases as well as the scope and content of the ABWR Standard Safety Analysis Report (SSAR). In Section 8.7, Water Chemistry Guidelines, of the referenced letter, it states that G.E. has committed to using BWR Owners Group water chemistry guidelines. These guidelines are necessary to maintain proper water chemistry in BWR cooling systems to prevent intergranular stress corrosion cracking of austenitic stainless steel piping and components and to minimize corrosion and erosion/corrosion-induced piping wall thinning in single-phase and two-phase high energy carbon steel piping. Water chemistry is also important for the minimization of plant radiation levels due to activated corrosion products. Section 10.4.6.3 of the ABWR indicates that the condensate cleanup system complies with Regulatory Guide 1.56. Section 10.4 should indicate that the system meets the guidelines published in:

> EPRI NP-4947-SR, BWR Hydrogen Water Chemistry Guidelines 1987 Revision, dated October 1988.
>
> EPRI NP-5283-SR-A, Guidelines for Permanent BWR Hydrogen Water Chemistry-1987 Revision, dated September 1987.

The use of zinc injection as a means of controlling BWR radiation-field build-up should be discussed.

281.16

In Section 10.4.6.3, the ABWR SSAR indicates that the condensate cleanup system removes some radioactive material, activated corrosion products and fission products that are carried over from the reactor. More important functions involve removal of condensate system corrosion products, and possible impurities from condenser leakage to assure meeting BWR Hydrogen Water Chemistry Guidelines. This should be discussed.

281.17

The condensate (Figure 10.4-4) and feedwater (Figure 10.4-7) system diagrams do not indicate the location of the oxygen injection into the condensate system and hydrogen and zinc oxide into the feedwater system. This information should be provided.

281.18

Section 10.4 does not discuss design improvements involving material selection, water chemistry, system temperatures, piping design and hydrodynamic conditions that are necessary to control erosion/corrosion. The EPRI CHECMATE or other erosion/corrosion computer codes may be useful design tools to minimize wall thinning due to erosion/corrosion-corrosion. The ABWR SSAR should discuss design considerations to minimize erosion/corrosion and

procedures and administrative controls to assure that the structural integrity of single-phase and two-phase high-energy carbon steel piping system is maintained.

430.59
Provide information on the following figures and tables: (10.1)

(1) Figure 10.1-2, Heat Balance for Guaranteed Reactor Rating

(2) Figure 10.1-3, Heat Balance for Valves-Wide-Open

(3) Table 10.1-1, Summary of Important Design Features and Performance Characteristics of the Steam and Power Conversion System, with regard to:

- Condensate pumps: total head (ft.) and motor hp.
- Low pressure heaters: Stage pressure (psia) and duty per shell (Btu/hr) for heaters Nos. 1,2,3, and 4.
- High pressure heaters: Stage pressure (psia) and duty per shell (Btu/hr) for heaters Nos. 5 and 6.
- Low pressure turbine exhaust pressure to condenser.

430.60
Specify the value for time “T” in Figure 10.2-2. (10.2)

430.61
Provide a description of the bulk hydrogen storage facility mentioned in Section 10.2.2.2. (10.2)

430.62
Provide a description of the speed control unit, the load control unit and the flow control unit of the electrohydraulic (EHC) system. Your description should include how they perform their intended functions. Clarify whether the EHC system will fully cut off steam at 103 percent of rated turbine speed. (10.2)

430.63
For the turbine overspeed protection system (described in Section 10.2.2.4), the SSAR referred to redundant electrical trip signals. Provide information on the power source associated with each of the trip circuits (10.2).

430.64
As presented in Section 10.2.2.4 of the ABWR SSAR, the closing time of the extraction nonreturn valves is less than 0.2 seconds, while it is 2 seconds at current BWR plants. Provide

additional information on the design of these valves that supports the difference between the above closing time values. (10.2)

430.65

Clarify whether at least one main stop valve, control valve, reheat stop valve and reheat intercept valve will be inspected at approximately 3-1/3 years by dismantling them, and whether visual and surface examinations will be conducted for the valve seats, disks and stems. (Note: The above is an acceptance criterion for SRP Section 10.2) (10.2)

430.66

Identify preoperational and startup tests of the turbine generator in accordance with Regulatory Guide 1.68, “Initial Test programs for Water Cooled Power Plants,” as an interface requirement. (10.2)

430.67

As stated in Section 10.3.2.1, “the four main steam lines are connected to a header upstream of the turbine stop valves....”. However, according to Figure 10.3-2a, the main steam header is located downstream of the turbine stop valves. Identify whether the statement or figure is in error and revise the item in error so that the SSAR is consistent. (10.3)

430.68

Provide information on the leakage detection system for steam leakage from the MSSS in the event of a steam line break. Also provide information on the stated “safety feature designed into the MSSS” that will prevent radiation exposures in excess of the limits of 10CFR Part 100 in the event of a break of a main steam line or any branch line (SSAR Section 10.3.3.) (10.3)

430.69

For the following items identified in SSAR Figure 10.3-1: (10.3)

(a) Deaerating steam to condenser

(b) Offgas system

(c) Steam jet air ejectors

(d) Turbine gland sealing system

(e) Reheater

(f) Main steam bypass

Provide the following information:

a. Maximum steam flow (lbs/hr)

b. Type of shut-off valve(s)

c. Size, quality, design code, closure time, actuation mechanism and associated motive power of the valve(s).

430.70

Provide information on the following items:(10.3)

(a) Analysis for steam hammer and relief valve discharge loads issues.

(b) Power source to the solenoid valves for the inboard and outboard main steam isolation valves.

(c) Location of seismic interface restraint (e.g., interface of which buildings?)

(d) Route which the main steam lines, including the branch lines, pass up to the turbine stop valves.

(e) Specific design features provided to protect safety related portions of the main steam supply system, including the main steam isolation valves, against externally and internally generated missiles and adverse natural phenomena such as floods, hurricanes and tornadoes.

430.71

Describe provisions for operation of the main condenser with leaking condenser tubes. (10.4.1)

430.72

Provide the permissible cooling water inleakage rate and the allowed time of operation with inleakage. (10.4.1)

430.73

Provide information on the following items:(10.4.1)

(a) Provisions incorporated into the main condenser to preclude component or tube failure due to steam blowdown from the turbine bypass system.

(b) Worst possible flood level in the applicable buildings due to complete failure of main condenser and provisions for protecting safety related equipment located in the buildings against such flooding (note that ABWR SSAR Section 3.4 does not discuss the turbine building).

430.74

Discuss how the components of the main condenser evacuation system (MCES) conform to the guidelines of Regulatory Guide 1.26, 1.33. and 1.123 with respect to quality group classification and quality assurance programs.(10.4.2)

430.75

Provide the design pressure and normal operational absolute pressure for the MCES components that could contain potentially explosive gas mixtures. (10.4.2)

430.76
Identify the radiation monitoring provisions for the mechanical vacuum pump exhaust. Is the exhaust filtered by charcoal absorber and HEPA filters prior to release? (10.4.2)

430.77
Identify the number, location and functions (i.e., recording and annunciating alarm) performed by the hydrogen analyzers. Clarify whether they can withstand a hydrogen detonation. (10.4.2)

430.78
Clarify whether the air ejectors are redundant in the sense that one of them is a standby. (10.4.2)

430.79
Identify the components and portions of the MCES that are designed to withstand a detonation in the system. (10.4.2)

430.80
Discuss how the design of the turbine gland sealing system (TGSS) conforms to the guidelines of Regulatory Guide 1.26 as it relates to the quality group classification for the system, and the Regulatory Guide 1.33 and 1.123 as they relate to the quality assurance program. (10.4.3)

430.81
Provide a description of the exhauster blower provided for the TGSS. (10.4.3)

430.82
ABWR SSAR Subsection 10.4.3.1.2 states that the TGSS exhausts the noncombustible gases to the turbine building equipment vent system, however, Subsection 10.4.3.3 states that the TGSS exhausts the noncombustibles gases eventually to the main vent. Clarify how the TGSS exhausts are monitored. Also, clarify whether the main vent mentioned above is the plant vent referred to in SSAR Section 11.5. (10.4.3)

430.83
What is the source for the auxiliary steam? Justify why an advanced design will use essentially radioactivity free auxiliary steam (see SSAR Section 10.4.3.2.2) as a backup sealing source rather than as normal sealing source. Note that the use of a process steam supply for sealing purpose can result in significant operational radioactivity releases. (10.4.3)

430.84
For turbine bypass system:(10.4.4)

(a) Provide figures which delineate the system and its components.

(b) Clarify whether the system includes pressure-reducer assemblies for the bypass valves to reduce steam pressure prior to steam discharge into the condenser.

430.85

For the circulating water system: (10.4.5)

(a) Describe the function of the waterbox fill and drain subsystem mentioned in ABWR Subsection 10.4.5.2.1. Also, describe the “makeup water” shown in SSAR Figure 10.4-3.

(b) Provide the worst possible flood levels that can occur in the applicable plant buildings as a result of circulating water system failure and indicate how safety-related equipment located in the buildings is protected against such flooding.

430.86

How is the remote manual motor-operated shutoff valve (gate valve F 282) powered? (10.4.7)

430.87

Describe the design features provided to protect the safety-related portion of the condensate and feedwater system from internally generated missiles.

430.88

Provide a summary of the analysis of a postulated high energy pipe break for the feedwater piping in the steam tunnel including the design features provided (e.g., pipe whip restraints) for preventing adverse effects resulting from pipe whip, jet impingement and flooding.

430.89

Provide information on the analysis that shows that the entire feedwater system piping can accommodate water hammer events and the means to prevent water hammer loads due to hydraulic transients (10.4.7)

430.90

Provide detailed information on the feedwater control valve and controller design, including the features that ensure the design will be stable and compatible with the system and imposed operating conditions. (10.4.7)

20.2.11 Chapter 11 Questions

281.13

Table 11.1-4 indicates that the N-16 concentration in the steam is four times the normal value when hydrogen water chemistry (HWC) is used. HWC tests conducted at BWRs have indicated that N-16 activities have increased in the range of 1.1 to 5 times the N-16 concentrations observed during normal water chemistry operations. What is the basis of the factor of four increase for the ABWR? Is it based on the model for predicting HWC that was reported in “U.S. Experience with Hydrogen Water Chemistry for Boiling Water Reactors,” R. L. Cowan, C. P. Ruiz and J. L. Simpson, April 1988? (11.1)

281.14

In Section 11.5.2.1.1, there is no discussion of a dual set point for the main steam radiation monitors (MSLRMs) when HWC is used. Below 20% power, the MSLRM set point is established to detect high radiation levels in the main steam lines and provide signals for reactor scram and MSIV closure to reduce the release of fission products to the environment in the event of a control rod drop accident. When hydrogen is injected into the feedwater at power levels above 20%, the MSLRMs may have to be reset due to the increased N-16 activity in the main steam line. (11.5.2)

430.154

All figures in ABWR SSAR Section 11.2 except Figures 11.2-2a and 11.2-2b are not legible. Provide enlarged-size legible versions of the figures. (11.2)

430.155

For each liquid radwaste subsystem, provide the available margin for processing surge flows by comparing the expected normal daily waste generation rate with the design flow rate for the limiting processing equipment. (11.2)

430.156

Provide specific information detailing how the liquid radwaste systems meet Regulatory Guide 1.143 guidelines (C.1.2.1 through C.1.2.5) and C.4.1 through C.4.5. Provide layout diagrams as necessary. Describe the indications provided to the operator that a transfer from one storage tank to another (the design basis states that upon high level signals, inputs are automatically routed to a parallel tank, ABWR SSAR Section 11.2.1.2.) has occurred.

430.157

Provide information on the following items for the liquid radwaste system:

430.157(1)

Reactor coolant activity (RCA) fraction for each substream of the low conductivity waste (LCW) and high conductivity waste (HCW) streams and the effective RCA fractions for the LCW and HCW streams. Integrate expected average daily liquid radwaste inputs due to generation of chemical wastes, ultrasonic resin cleaning, cleanup phase separator decant

backwash, and unique design features of the ABWR with other inputs applicable to the LCW and HCW streams. Provide the total expected average daily input to the LCW and HCW streams. Note that ABWR SSAR Table 11.2-3 is incomplete. Some of the values for wastes generated given in the table are significantly lower than those given in the report ANSI/ANS-55.6 or NUREG-0016, Rev. 1. The values given in these reports for BWRs may be used but some minor adjustments for the ABWR design may be necessary (for example, drywell equipment drain input of 3400 gallons per day (gpd) which includes 2200 gpd due to recirculation pump seal leakage may require adjustment for the ABWR design). Also, define the terms CUW and CF mentioned in Table 11.2-3.

430.157(2)

Holdup times associated with collection and processing of the LCW and HCW streams; holdup time associated with discharge of the HCW stream.

430.157(3)

Capacities of all tanks, including sample tanks (in gallons) and processing equipment (in gpm) considered in calculating holdup times for the LCW and HCW streams. Include applicable discharge pump flow rate. State whether or not sample tanks are shared.

430.157(4)

Clarify how the liquid radwaste system has adequate margin to preclude liquid radwaste discharge even under a wide variety of anticipated operational occurrences.

430.158

For the detergent waste subsystems, provide the capacities of all tanks, flow rates of processing equipment and pumps. Clarify rain sample tanks. (ABWR SSAR Section 11.2.1.2 states that these wastes are discharged from the hot shower drain receiver tank whereas Section 11.2.3.1 states that these are discharged from the shower drain sample tanks.) Also, clarify whether the hot shower drain receiver tank has adequate capacity to collect the high volume of detergent wastes (31.3 cubic meters per day —Table 11.2-3) and whether storm drain(s) is also an input to the tank (ABWR SSAR Section 11.2.2.3 does not include storm drain; however, Table 11.2-3 shows a volume input of 20 meters/day from this source). (11.2)

430.159

ABWM SSAR Tables 11.2-4 and 11.2-5 do not indicate that detergent wastes, all of which are expected to be released untreated (note that the staff does not give any credit for radioactivity removal due to processing through a detergent filter), have been included in the table. NUREG-0016, Rev 1 has calculated a total of 0.09 ci/yr for the untreated release of detergent wastes. Revise these tables to include the untreated release of detergent wastes. Also, include the expected tritium release via the liquid effluents. Further, provide the basis for releasing the detergent waste via the liquid pathway untreated. (11.2)

430.160

Limiting value of 200 gpm for discharge of liquid radwaste to the discharge canal in conjunction with minimum dilution volume of 1500 gpm gives only a low minimum dilution factor of 7.5 for a critical liquid pathway exposure i.e., fishing in the discharge canal. The above compares with a dilution factor of 200 and much above that quoted by a number of operating BWRs in the periodic effluent reports. With the expected release of high conductivity liquid wastes (can be up to 10 percent of the total high conductivity wastes) and untreated detergent wastes, it is not clear whether the dilution factor of 7.5 will be adequate to ensure compliance with 10CFR50, Appendix I dose limits for liquid pathways, even if an additional dilution factor of 5 is included between the canal and subsequent consumption or recreational activity involving liquid effluent. Also, it is not clear whether the low dilution factor will make it difficult to complete monitoring prior to release of liquid radwaste. Address the above concerns by either decreasing the limiting value of liquid radwaste discharge rate or increasing the minimum dilution volume or doing both of the above. (11.2)

430.161

Clarify whether the seismic Category I steel-lined radwaste building substructure (see response to Question No. 430.58 dated March 7, 1989) includes the base mat and outside walls to a height sufficient to contain the maximum liquid inventory expected to be in the building. (11.2)

430.162

Regulatory Guide 1.143, Position C.2, provides that, for a system with a design pressure of less than 1.5 atmosphere absolute, the supports for the charcoal tanks and the buildings housing there tanks meet the seismic design criteria of Position C.5. Clarify whether the charcoal adsorber vault meets these requirements. Include a discussion of how the gaseous waste management system meets Position C.5 guidance. (11.3)

430.163

Figures 11.3-1 and 11.3-2 have been reduced so that portions of each of these figures are not legible. Please provide legible versions of these figures. (11.3)

430.164

The combination of the design dewpoint (30 F, ABWR SSAR Section 11.3.4.2.7), the system operating temperature (100 F, ABWR SSAR Section 11.3.3.1) and the mass of charcoal (12 tons, ABWR SSAR Table 11.3-2) gives significantly lower dynamic adsorption coefficients for krypton and xenon, and consequently much lower holdup times for these gases in the offgas processing system than those given in ABWR SSAR Section 11.3.2. Correct the values for adsorption coefficients and the holdup times as appropriate. Note that the staff calculates the holdup times using the expression given in NUREG-0016, Rev. 1 (page 2-35). Also, note that above parametric values will result in substantially higher noble gas releases to the environment (e.g., about 10^5 Ci/yr for Xe-133) from the offgas treatment system. Provide the dynamic adsorption coefficient and holdup time for Argon-41 also. [In this context, the staff notes that GE provided the same holdup times for xenon and krypton, i.e., 42 days and 46 hours

respectively, in the SAR for GESSAR-II, which uses a refrigerated charcoal delay bed system containing 24.6 tons of activated charcoal.] Additionally, justify the apparent significant reduction in holdup time for noble gases for an advanced design such as ABWR.

430.165

Describe provisions to control leakage paths to the environment after a hydrogen detonation within the gaseous waste management system.

430.166

Provide information on the following items for the gaseous waste management system:

430.166A

Hydrogen concentration instrumentation and associated alarm provisions. Discuss how the ABWR instrumentation conforms with applicable guidelines of SRP 11.3, Acceptance Criterion II.B.6, pages 11.3-4 through 11.3-6. Also, discuss how the offgas system design complies with GDC 3 as it relates to providing protection to the system from the effects of an explosive mixture of hydrogen and oxygen.

430.166B

Holdup time for off-gasses from the main condenser air ejector off-gas treatment system. The staff notes that GE did not provide a satisfactory response to the above question raised earlier (see GE's response dated March 7, 1989 to Question No. 460.4.7).

430.166C

Offgas system alarmed process parameters (provide in tabular form).

430.166D

Design holdup time for gas vented from the gland seal condenser, iodine partition factor for the condenser, and fraction of radioiodine released through the system vent. Provide expected annual noble gas and iodine releases to the environment (including the basis and rationale) from the turbine gland seal system resulting from use of steam generated from main steam and high pressure heater drain tanks for sealing the turbine gland (see GE'S response to Question No. 430.83 dated February 28, 1990.)

430.166E

Provisions incorporated to reduce radioactivity releases through the ventilation systems (turbine building, etc.) (e.g., HEPA filter, charcoal adsorbers and their thickness). Discuss how the ABWR systems conform with the guidelines of Regulatory Guide 1.140, with respect to the treatment systems for these release paths.

430.166F

Release points, effluent flow rates through them and their other characteristics (see NUREG-0016, Rev. 1, Section 4.7, Item 4).

430.166G
Provide a discussion on compliance with GDCs 60 and 64 for all gaseous releases to the environment (do not limit this discussion to the offgas system).

430.166H
Monitoring of the individual performance of the equipment within the offgas system.

430.167
The total annual noble gas release from the offgas treatment system given in ABWR SSAR Table 11.3-1 is incorrect. Also, the table lists only releases from the offgas treatment system. Provide a table listing expected annual total airborne release from all sources (offgas system, mechanical vacuum pump, gland seal, building ventilation releases including containment purges) for noble gases including Argon-41, iodines, particulates, carbon-14, and tritium during normal plant operation including anticipated operational occurrences. (11.3)

430.168
Provide a table comparing airborne effluent concentrations for all radionuclides during periods of fission product release at design levels from the fuel with 10 CFR Part 20 concentration limits (11.3.)

430.169
Section 11.4.2.3.5 makes reference to storage of containers until they can be shipped. However, no description of the storage facility for solid wastes is provided. Provide information regarding these storage facilities that shows that these facilities will meet the guidance of BTP ETSB 11-3, Part B.III and Regulatory Guide 1.143.

430.170
Provide a table of expected waste volumes generated annually by each “wet” solid waste source (normal and greater-than-expected surge waste volumes) and the capacities of all tanks accumulating spent resins and filter sludges. Provide the corresponding specific activity for each “wet” solid waste source. These tanks should be sized so as to meet the storage requirements of BTP ETSB 11.3, Part B.III.l. Provide an estimate of expected annual “dry” solid wastes and the corresponding curie content.

430.171
Table 11.4.2 shows a solid waste generation rate of 97.3 m^3/y. This amount of solid waste is significantly smaller (by an order of magnitude) than values used in previous FSARs for BWRs. Provide the justification for this reduction in waste production or revise the estimate for waste production rates.

430.172
Identify which of the design criteria from ABWR SSAR Section 11.2.1.2 are deemed to be applicable to the solid waste system. Specifically, the criteria of Sections 11.2.1.2.1 (Quality

Classification, Construction and Testing Requirements) and 11.2.1.2.2 (Seismic Design) should be applicable to the solid waste system and should be identified in Section 11.4.

430.173

Verify that the structures containing the solid waste system meet the seismic qualifications of Regulatory Guide 1.143, Position C.3.13. (11.4.)

430.174

Section 11.4.1.2 (Design Criteria) states in part that “Proportional amounts of waste and fixative are incorporated to insure that no free water accumulates in the waste container.” Provide details on the procedures needed to insure proper mixing and to detect free water if still present after mixing.

430.175

Discuss compliance of the solid waste management system with 10 CFR Section 20.106, GDCs 60 and 63 requirements. Include both “wet” solid wastes and “dry” solid wastes in the discussion. (11.4.)

430.176

Discuss how the solid waste management system meets the guidelines identified under “Additional Design Features” in BTP ETSB 11.3, Part B.V.

460.1

With respect to radioactive source terms and the calculations of subsequent release to the environment, discuss your position in terms of the regulatory guidance provided in NUREG-0800, SRP 11.1, such as NUREG-0016, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors,” Revision 1 and Regulatory Guide 1.112, “Calculation of Release of Radioactive materials in Gaseous and Liquid Effluent from Light-Water-Cooled Power Reactors.” (11.1)

460.2

Clarify whether the radioactive source terms given in ABWR SSAR Tables 11.1-1 through 11.1-5 have been adjusted to the maximum core thermal power of the ABWR evaluated for safety consideration in the SSAR. (11.1)

460.3

Check and correct as appropriate the following: (11.1)

1. Caption for Column 3 of SSAR Table 11.1-1
2. Kr-87 value given in Column 4 of SSAR Table 11.1-1
3. N-16 steam and reactor water concentrations given in SSAR Table 11.1-4.

460.4

The staff requires the values of some parameters for performing an independent evaluation of the ABWR Reactor Coolant System (RCS) radioactive source terms. These are used in conjunction with radwaste management systems applicable for specific plants referencing the ABWR to determine the adequacy of the specific radwaste management systems (see NUREG-0016, Rev. 1 Chapter 4). Therefore, provide information on the following parameters or provisions: (11.1)

1. Thermal Power (Mwt)
2. Total steam flow rate (lb/hr)
3. Mass of water in the RCS (lbs)
4. Steam/water concentration ratio, i.e., reactor vessel carry over factor for halogens and particulates
5. Main condenser tubing material of construction (stainless steel or copper)
6. Powdex or deep bed condensate treatment
7. Air ejector offgas holdup time (hr)
8. Charcoal delay system for treating offgases:
    a. Operating and dew point temperatures of the delay system
    b. Mass of Charcoal (lbs)
    c. Dynamic absorption coefficients (cm^3/g) for Kr, Xe and Ar
9. Clean or radioactive steam for gland seal
10. Mechanical vacuum pump iodine release fraction if within the ABWR scope
11. Provisions incorporated to reduce radioactivity releases through the ventilation or exhaust systems that come within the ABWR scope (e.g., HEPA filters, charcoal adsorbers and their thickness)
12. Release points characteristics (see NUREG-0016, Rev. 1, Chapter 4, Section 4.7, Item 4). Include description of the main stack.

Note that Item Nos. 5 and 6 are required to determine the carry over factor for radiohalogens from reactor water to steam. Also, note that when a summary of building ventilation system and mechanical vacuum pump releases are provided in December 1988 (see SSAR Section 12.3.3), additional information on the releases may be requested.

460.5

Regarding Process and effluent Radiological Monitoring and Sampling Systems: (11.5)

1. Provide locations of plant vent, radwaste building vent(s), offgas exhaust vent, turbine building vent and all other exhaust vents through which all radioactive gaseous or airborne effluents are discharged directly to the environment. Also, provide individual gaseous or airborne radioactive effluents (i.e., drywell purge, release via SGTS, release from RCIC, RWCU and ECCS equipment rooms, release from shield wall annulus, fuel area, battery rooms, CRD maintenance area, release from any other secondary containment area not listed above, release from mechanical vacuum pump, radwaste building control room and unit substation, release of treated offgases, release from turbine building, and any other effluent not listed above) discharged through each one of the vents directly to the environment.

2. Provide the locations of all the process monitors for radioactive gaseous or airborne effluents (i.e., containment HVAC radiation monitors, fuel area ventilation exhaust monitors, battery room and CRD maintenance area radiation monitors, shield wall annulus monitors, RCIC, RWCU and ECCS equipment rooms radiation monitors, any other secondary containment area radiation monitors not listed above, radwaste building control room and unit substation radiation monitors, and any other process monitor not listed above). Include all the process radiation monitors not currently listed in SSAR Section 11 Tables (e.g., shield wall annulus monitors, RCIC, RWCU and ECCS equipment rooms monitors, other secondary containment area radiation monitors).

3. Clarify which areas process monitors other than the fuel area ventilation exhaust monitors will initiate the startup of SGTS on detection of high airborne radioactivity level in the area (e.g., ECCS/RWCU/RCIC equipment rooms, shield wall annulus, primary containment purge).

4. Clarify whether all gaseous or airborne radioactive effluents from the plant are monitored and which among them are continuously monitored.

5. SSAR Tables 11.5-1 and 11.5-2 refer to plant vent discharge and plant vent elevated discharge respectively. Clarify whether ABWR has two different plant discharge vents. If so, provide the location of the plant vent for elevated discharge and the radioactive gaseous or airborne effluent discharge via that vent.

6. Explain why ABWR design has only one channel for offgas post-treatment monitoring (see SSAR Table 11.5-1).

7. Discuss how ABWR design complies with the requirements of NUREG-0737, “Clarification of TMI Requirements,” Item II.F.1, Attachments 1 and 2 with regard to monitoring instrumentation for noble gases and sampling and analysis of plant effluents for radioiodine, during accident conditions.

8. Clarify how the detergent and chemical wastes releases are monitored.

9. Clarify the references to service water effluent, essential service water system-RHR, and component cooling water system made in SSAR Section 11.5 and the associated tables (this question arises since the above nomenclatures have not been used for the ABWR water systems-see SSAR Section 9). Correct the entries in SSAR Section 11.5 and the Tables 11.5-1, 2 and 3 as appropriate.

10. Clarify whether ABWR design requires two condensate storage tanks (see SSAR Table 11.5-4).

11. Identify all the interface requirements (e.g., expected activities, alarms and trips for a number of monitors (see SSAR Table 11.5-2), monitoring mechanical vacuum pump and turbine building exhausts, minimum dilution required for liquid radwaste effluent).

20.2.12 Chapter 12 Questions

471.1

Section 12.1.1.2 of the submittal states that operational policies are out of the Nuclear Island scope. Following Section 12.1.1.3, the report states that "Compliance of the Nuclear Island design with Title 10 of the code of Federal Regulations Part 20 (10 CFR 20), is ensured by the compliance of the design and operation of the facility within the guidelines of Regulatory Guides (RG) 8.8, 8.10, and 1.8." Further, Section 12.1.1.3.2 states that R,G. 8.10 is out of the Nuclear Island scope and Section 12.1.1.3.3 states that R.G. 1.8 is out of the Nuclear Island scope. The applicant should clarify these statements by describing to what degree the guidance in R.G. 8.8, 8.10, and 1.8 is incorporated into the ABWR design.

471.2

Section 12.1.2.2.1 indicates that in lieu of specific instructions, design engineers were instructed to incorporate the applicable design criteria in R.G. 8.8. What mechanism in the design process ensured that <u>all</u> applicable criteria were considered in the individual design.

471.3

Section 12.1.2.2.2, paragraph (4) states "Past experience has been factored into current designs. The steam relief valves have been redesigned as a result of inservice testing. Access for inservice inspection has been changed." State in what respect the access has been changed, and what is the impact of this change on occupational radiation exposure (ORE).

471.4

Section 12.1.2.3, paragraph (4), last sentence is not completed. It states: "These systems are designed to limit the radioactive." Complete the sentence.

471.5

Section 12.1.2.3 should address the reduction of personnel exposure due to the elimination of external primary coolant recirculation loops in ABWR design.

471.6

Section 12.1.2.3.2, paragraph (5) refers to "packaged units". State whether or not this includes skid mounted components; if not, define what is meant by "packaged units".

471.7

Section 12.1.2.3.2, paragraph (8) refers to providing means for decontamination of service areas; clarify the statement by providing examples of means of decontamination and of service areas referred to.

471.8

Table 12.2-3 part A, shows gamma ray sources in the core during operation (MeV/sec-W) for various energy bounds (MeV). Provide the basis for these data.

471.9
Table 11.1-4, Coolant Activation Products in Reactor Water and Steam, indicates that values in steam for N-13, N-16 and N-17 should be multiplied by a factor of four when hydrogen water chemistry is used. Explain why the concentrations of these isotopes remain unaffected in reactor water when hydrogen water chemistry is used. State whether the effect of hydrogen water chemistry on the nitrogen activation product concentration (increase by a factor of four) was incorporated into the plant shielding design.

471.10
Address in Section 12.1.2.2.3 the selection of materials and instrumentation with respect to radiation exposure damage, frequency of maintenance, and ALARA personnel radiation exposure (i.e., reactor coolant pump component and material selection).

471.11
Provide drawings of cross sections R4 (0^o-180^o) and RD (90^o-270^o) of the ABWR reactor building for better orientation of radiation zones.

471.12
In Table 11.1-1, Noble Radiogas Source Terms (Steam), (page 11.1-9) revise the description of first column: Source Term (t=30 min) to Source Term (t=0 min).

471.13
In accordance with Section 12.2.2 of R.G. 1.70, provide a description of radioactive sources in the spent fuel pool. This description should include the expected radioactive concentrations in the spent fuel pool water, as well as contained sources within the pool.

471.14
In accordance with R.G. 1.70, Section 12.2.2, Airborne Radioactive Material Sources, provide average expected annual airborne concentrations, at normal operating and anticipated operational occurrences, in various areas of the plant normally occupied by operating personnel.

471.15
Provide the expected N-16 source strength increase in steam leaving ABWR pressure vessel due to the elimination of the external reactor coolant loops in comparison to BWR plant with external coolant loops of the same power level. Provide the method of calculation.

471.16
Address TMI issues in accordance with NUREG-0737 as it relates to ABWR design Section 12.

471.17
Provide the missing Table 12.2-1,A,B,C,D; Table 12.2-3,C; and Table 12.2-4,B: Tables 12.2-5 through Tables 12.2-21; Section 12.3.3; Section 12.3.4; Table 12.3-3;
Figures 12.3-8 through 12.3-23; and Section 12.4.

471.18
Describe the radiological impact of each of the advanced design features of the ABWR design. Show how ALARA considerations were engineered into these features by describing the source term, reliability, maintenance and surveillance associated with these components. Features discussed should include, but not be limited to, the internal reactor circulating pumps, the control rod drive mechanisms, hydrogen water chemistry and reactor vessel bottom head design.

471.19
Provide scaled layout and arrangement drawings of the facility showing the location at all sources described in Chapter 11 and Section 12.2, for the entire nuclear island, including inside the drywell. Layouts should show major shield wall thicknesses, controlled access areas, personnel and equipment decontamination areas, contamination control areas, location at airborne radioactivity and area radioactivity monitors, location of Health Physics facilities, post-accident sampling station and counting room.

471.20
Specify the design basis radiation level in the counting room during normal operation and anticipated occurrences.

471.21
Provide plant layout drawings that detail the radiation zone boundaries for refueling outage and accident conditions, in addition to the normal operating conditions shown on Tables 12.3-1 through 12.3-7 of the submittal. Layouts should show access control features and traffic patterns for the entire nuclear island.

471.22
Identify all plant areas where radiation levels of 100 rads/hr, or more, could result from normal operations or anticipated operational occurrences and describe additional control measures to protect workers from these hazards.

471.23
The last paragraph in Section 12.3.1.3 states that all areas with radiation levels greater than 100 mRem/hr will be locked. This implies that the ABWR design will not incorporate Standard Technical Specification 6.12 which allows areas to remain unlocked up to 1000 mRem/hr. If this is not the case the radiation zone maps should be revised to identify zones with radiation levels greater than 1000 mRem/hr.

471.24
Figures 12.3-3 and 12.3-4 show two stairwells on the east side of valve rooms B and C where the access to and from the stairs is in a low radiation zone (< 5 mRem/hr), but the stairwell itself is in a high radiation area (≥ 100 mRem/hr). Justify why additional shielding in the stairwell to prevent an unwarranted high radiation traffic area is not reasonably achievable.

471.25
Figure 12.3-2 also shows an arrangement where one would have to traverse a high radiation area to get to a low radiation area. The insert diagram in Figure 12.3-2 shows that access to the TIP drive room (< 5 mRem/hr) is through a room on the south side which is a high radiation area. Justify the planned access to the TIP drive room. Also provide the zoning layout for the entire 1500 mm level.

471.26
The first paragraph at the top of page 12.3-13 states that in the event of a complete TIP retraction, egress from the TIP room is possible with less than 100mR radiation exposure. What features has the ABWR design incorporated to ensure that exposures received from the recovery from this event are ALARA?

471.27
Page 12.3-13 of the submittal has a statement that a concrete CRD storage vault, used for storing CRD parts and assembled units, is provided in the CRD maintenance room. This design feature is not indicated in Figure 12.3-2. Provide a figure depicting this design feature. Discuss the anticipated source term within this vault and associated shielding requirements.

471.28
Provide information on the shielding for each of the radiation sources identified in Chapter 11 and Section 12.2, including the criteria for penetrations, the material, the method by which the shield parameters (cross section, buildup factor, etc.) were determined and the assumptions, codes, and techniques used in the calculations. Describe how the guidance provided in R.G. 8.8 has been followed in special protection features.

471.29
Table 12.3-1 lists five computer shielding codes used in the ABWR design. The last entry in the table states “Additional Codes to be added by Applicant”. Identify these codes and give a full description of their application to the ABWR design, or clarify the use of the term “Applicant”.

471.30
Describe whether the concrete shielding of the ABWR design follows the guidance on fabrication and installation in R.G. 1.69 in all cases. If not, describe the specific alternative method used.

471.31
Sections 12.2.2.1 and 12.3.2.2.2 state that the ABWR shielding design is based on a fission product release rate of 50,000 mCi/sec of noble gas after a 30 minute decay time. The standard assumption (see Standard Review Plan p.12.2-4) is 100,000 mCi/sec. Justify the use of this much lower source term.

471.32

Describe any temporary shielding required to assure protection of individuals present in the upper drywell during refueling and fuel transfer operations. The description should include shield thickness, and material required for both normal and anticipated operational occurrences.

471.33

The acceptance criteria for radiation streaming through reactor shield wall penetrations (on page 12.3-12) is unclear. Describe the radiation streaming through reactor shield wall penetrations during refueling operations for all feasible fuel configurations.

471.34

Section 12.3.5 of the submittal identifies areas requiring access to mitigate the consequences of an accident. Indicate whether this is a complete list of the vital areas (as described in item II B.2 of NUREG-0737) of the facility. If not, identify the vital areas of the facility; and if so, justify why the post accident sampling station (PASS) and the counting rooms are not considered vital.

471.35

Provide a description of the design features needed to assure adequate access to vital areas. The description should identify major sources of radiation considered and protective requirements (for example the response should address the contribution to radiation exposure at the PASS location from the stack monitor room).

471.36

Between pages 12.3-6 and 12.3-12, it appears that the designators RWSC, RWCU and CUW are all being used interchangeably to refer to the Reactor Water Clean Up System. Verify which is correct and delete or define the other acronyms used.

471.37

The fourth paragraph at page 12.3-9 indicates that drains from the SGTS filter housing will be piped directly to a floor drain sump. Industry experience has shown that these housing drains can provide bypass pathways around filter/absorber beds. Provide a description of the ABWR SGTS and Control Room filter housing drains showing how filter bypass is prevented.

471.38

Section 12.3.1.1(3) states that connections are provided for decontamination at heat exchangers in “highly radioactive systems.” Identify these systems and heat exchangers.

471.39

Provide layout drawings of the control room showing radiation zones during normal operation, anticipated operational occurrences, and design basis accidents. Shield wall thickness, calculational parameters (and assumptions), and the models used to determine compliance with GDC 19 should be indicated.

471.40

Figure 12.3-7 indicates that the area above the spent fuel pool is a high radiation zone. Provide dose calculations for refueling and other anticipated operations above the pool. Calculations should include contribution from activity suspended in pool water as well as direct radiation from spent fuel and other components in the pool.

471.41

Figure 12.3-6 shows several small A zones (0.6 mRem/hr) completely surrounded by higher level zones (C zones). What are the purposes of these areas and justify why continuous A zones cannot be provided.

471.42

EPRI-ALWR Requirements Document in Chapter 12: Radioactive Waste Processing System (page 12.3-8), Section 3.2.1, Goal for GRWPS Radioactivity Releases state that:

> "The total radioactivity of gases released from the paint (excluding the activity of released tritium) shall not exceed the following values: BWR 2,000 curies per year, etc."

ALWR, standard plant, in Chapter 12, Radiation Protection, in Section 12.2.2.1, Production of Airborne Sources, last paragraph states that:

> "Approximately 7,900 Ci/plant/year of noble radiogases are released; one-half of this is released from the turbine building....."

Please address this apparent discrepancy.

20.2.13 Chapter 13 Questions

910.7

The ABWR Licensing Review Bases document states in its section 7.1 that the importance of such potential contributors to severe accident risk as sabotage should be carefully analyzed and considered in the design of new plants. To permit our review of this analysis and considerations, please provide a discussion of the insider and outsider sabotage actions that would be necessary to cause significant core damage or Part 100 release levels. This discussion should include identification of the ABWR design features that decrease reliance on physical security programs for sabotage protection. (13.6.1)

910.8

The terminology used should not differ from terminology used in 10CFR73, such as “high-security areas” instead of “protected and vital areas”, and “clear areas adjacent to the physical barriers” instead of “isolation zones.” (13.6.3.7)

910.9

At many current generation BWRs, protection of essential service water systems, needed for support of emergency diesel generators and for suppression pool cooling, are among the most demanding security system requirements because of their components' locations at the peripheries of the protected areas. For the ABWR, these cooling functions may be performed by the reactor building cooling water system (RCW). If it is not intended that RCW be designated as a vital system, please provide justification for that position. If it is intended that RCW be designated as a vital system, please provide sabotage protection interface requirements for the RCW and any supporting systems required for it to support a safe reactor shutdown. In addition to protection of safety-related portions of the RCW system, these interface criteria should address prevention of sabotage of any nonsafety-related parts of the RCW system from interfering with its safety-related functions. They also should address criteria on portions of the system, such as the sea water heat exchanger building, that may be site specific and not in the Nuclear Island scope. (9.2.11, 9.2.16 & 13.6.3)

910.10

In addition to measures for detection of the inoperability of vital equipment and for control of access to areas containing vital equipment, site specific requirements for certain security systems (e.g., uninterruptible security lighting and power, and in-plant security communications) might be more efficiently incorporated and avoid impacts on ABWR design scope safety systems if considered in the standard plant design stage. Discuss what provisions for these security systems have been provided in the standard design, and provide interface criteria that will allow the security requirements for these systems to be accomplished without adversely interfering with safety systems. Include criteria necessary to assure that:

(a) There are no areas within the Nuclear Island where communication with the central and secondary alarm stations is not possible;

(b) Portable security radios will not interfere with plant monitoring equipment;

(c) Minimum isolation zone and protected area illumination capabilities cannot be defeated by sabotage actions outside of the protected area; and,

(d) Electromagnetic interference from plant equipment startups or power transfers will not create nuisance alarms or trip security access control systems. (9.5.2, 9.5.3 & 13.6.3.7)

910.11

Submit the analysis that supports the vital areas results described in this section. Affirm that these areas include all of the reactor coolant pressure boundary, including appropriate motor control centers and power supplies, and systems required for mitigation of transients, and support systems (e.g., cooling water, instrumentation, control power) necessary for these systems to operate. Delineate which systems are included in paragraph (1)(a) of Section 13.6.3.3 as vital “core cooling systems,” and which components in these system are vital components. Which vital systems would be out of the scope of the standard Nuclear Island and thus subject to plant specific review? (13.6.3.3)

910.12

Localized alarmed doors and keyed cylinder lock doors are not acceptable for control of access to vital areas. 10CFR73.55(e) requires all vital area alarms to indicate in two alarm stations. #73.55(d)(7) requires all unoccupied vital areas to be locked and alarmed. It also requires provisions for rapid ingress or egress. All doors and hatches into or out of vital areas should have balanced magnetic switches with tamper-safe cabling. (13.6.3.4)

910.13

The effectiveness of grills and grates, used to prevent access through ducts and air intake and exhaust systems, may depend on how isolated and hidden from view is the exterior of the vital area barrier at the opening under consideration. Steel grills and 3/4-in. steel bars can be cut with hand tools and may not provide sufficient protection in isolated locations. Regulatory Guide 5.65 provides some examples of alternative ventilation barriers with longer penetration times. Furthermore, ducting to the control room should also satisfy the bullet resistant requirement of 10CFR73.55(c)(6) for protection of the operators in the control room. (13.6.3.6)

910.14

To prevent confusion with national security information usage of the term “confidential”, please follow the directions in NUREG-0794, Protection of Unclassified Safeguards Information, for appropriate marking, and handling, of sensitive but unclassified safeguards information. (13.6.3)

910.16

Replace references to “industrial sabotage” with “radiological sabotage”, as defined in 10CFR73.2(p). (13.6)

910.17

Response 910.10 stated that in-plant security communications requirements of 10CFR73.55(f) are outside the scope of the ABWR Standard Design and would be the responsibility of the certification users. NRC Information Notice 83-83, '"Use of Portable Radio Transmitters Inside Nuclear Power Plants," discussed concerns about the potential for radio frequency interference (RFI) from portable radio transmitters to cause reactor system malfunctions and spurious actuations. A capability for continuous communication between security personnel on patrol within vital areas of the plant and the security alarm stations is required by 73.55(f). Common practice is to use hand held radios to meet this requirement. As noted in Information Notice 83-83, administrative prohibitions on the use of portable radios in certain areas of the plant may not adequately resolve the concern, particularly for new designs that make extensive use of solid state devices in instrumentation and control circuits. The ABWR Licensing Review Basis (August, 1987) stated that the ABWR SSAR will not provide details but will identify design requirements for 73.55(f)(1). Please address design requirements to assure that means can be provided for continuous communication between security personnel stationed within, or on patrol within, vital areas of the plant and the security alarm stations, without interference with plant instrumentation and control.

910.18

Generic Letter 87-08 states that an uninterruptible power supply is preferred for alarm annunciator equipment and non-portable communications equipment. Industry standard ANSI/ANS-3.3-1988 states that intrusion detection aids (e.g., door alarms, fence alarms, and the alarm assessment [closed circuit television] system) should also be supplied with uninterruptible power. Regulatory Guide 5.65 notes that an uninterruptible power supply for electrical locking devices on vital area doors is an acceptable method for providing the prompt access to vital equipment required by 10CFR73.55(d)(7)(ii). Section 8.3 of the ABWR SSAR discusses onsite power systems, including non-class 1E vital AC power for important non-safety related loads, but makes no mention of security system power requirements. The draft EPRI-ALWR Requirements Document quoted in ABWR SSAR Appendix 19B says that the security power subsystem shall be a non-interruptible power source. Therefore, we again request you to discuss what provisions for these security systems have been provided in the standard design, and provide interface criteria that will allow the security requirements for these systems to be accomplished without adversely interfering with safety systems.

910.19

Explain why the environmental conditions parameters of ABWR SSAR Appendix 3I should not apply to the design and qualification of security access control systems. Consider desirability of operable card reader controlled door locks in the event of a pipe break, such as occurred at Surry (NRC Augmented Inspection Team Report 50-281/86-42).

910.20

The list of vital areas and vital equipment in Subsection 13.6.3.3 appears to include all of the reactor coolant pressure boundary, including appropriate motor control centers and power

supplies; systems required for mitigation of transients; and support systems (e.g., cooling water, instrumentation, control power) necessary for these systems to operate; as well as other safety related systems. Are there any exceptions to this statement?

910.21

Subsection 13.6.3.4 still specifies door alarms only for doors at card reader locations. All doors and hatches connecting vital areas to non-vital areas should be alarmed (e.g., balanced magnetic switches with tamper-safe cabling), not just doors at card reader locations, with the alarm hardware being on the vital side of the door.

910.22

10CFR73.55(d)(7) also requires provisions to accommodate the potential need for rapid ingress or egress. Emergency exits should include provisions for exiting without use of keys or card readers. Please include appropriate language in Subsection 13.6.3.4.

910.23

Certain rooms are identified in Subsection 13.6.3.6, Bullet-Resisting Walls and Doors, Security Grills and Screens, as:

> “... a particularly high security zone. Specific precautionary measures have been incorporated into the building design to minimize forcible access to this area.”

This seems to confuse two requirements of 10CFR73.55. Bullet-resisting barriers are required by 10CFR73.55(c)(6) for the control room. According to 10CFR73.55(c)(1), access to all the vital areas identified in Subsection 13.6.3.3 requires passage through two physical barriers of sufficient strength to meet the performance requirements of 10CFR73.55(a). As noted in Regulatory Guide 5.65, a vital area barrier is to be constructed of materials that provide delay to forcible access from non-vital areas.

910.24

The change made to Subsection 13.6.3.6 for Response 910.13 is too vague. Will the design of air exhausts, HVAC gratings, and other man-sized (i.e., 96 square inches) openings, in all physical barriers that separate vital areas from non-vital areas, satisfy the criterion in NUREG-0908 and Regulatory Guide 5.65 that the integrity of a vital area barrier containing them not be decreased?

910.25

The meanings of some statements in Section 13.6.3 are unclear and maybe unnecessary.

(a) Subsection 13.6.3.1, Introduction, includes “the capability for detection of inoperability of vital equipment” as a concern of the physical security design requirements. Is this what it was meant to say? This is not typically a physical security function. What portions of Chapter 7 discusses this?

(b) What is the intent of the last sentence of Subsection 13.6.3.3: “Hence, access control is considered separately.”?

(c) The interface requirements of Subsection 13.6.3.7, Compatibility with the Remainder of the Plant, would be covered in the site security plans required by 10CFR50.34(c) and (d). Section 13.6.2 already states that the security plans are out of the scope of the ABWR Standard Plant design certification which means they would be required to be provided by applicants referencing the certified design. Of the eleven items listed, only #3 appears to be a unique ABWR interface requirement.

If 13.6.3.7 is intended to clarify what additional security requirements those applicants would need to satisfy, the list is incomplete, as it omits lighting and other requirements.

20.2.14 Chapter 14 Questions

None.

20.2.15 Chapter 15 Questions

420.28

Section 15.A.2.2 defines “Safety” and “Power Generation.” The staff did not locate definitions for “important to safety” and “safety related” which are used in Chapter 7. (15A)

420.96

The safety system auxiliaries (Figure 15A.6-1) should be modified to include any HVAC required to assure continued operation of the electronics. (15A.6)

420.118

Describe when appropriate operator action in seconds is required to prevent significant radiological impact. (15.2.4.5.1)

420.122

Is the instrumentation required for the operator to verify bypass valve performance and relief valve operation 1E or N-1E? (15.2.2.2.1.4)

420.123

SSAR 15B.4 describes the essential multiplexing system (EMS) in some detail. SSAR Figure 7A.2-1 states that the design is not limited to this configuration. It is our understanding that the EMS design is still in a preliminary design stage. Is SSAR 15B.4 still accurate and is the design limited to that configuration? (15B4)

420.124

The FMEA submitted in SSAR 15B.4 is inadequate for a safety evaluation supporting the design certification. The FMEA appears to the staff to be oversimplified with one line item each for component failure and does not address potential software complications. The staff requests clarification of how this FMEA was developed given that the system design has not been finalized. The staff also believes that software failures need to be evaluated. The failure modes investigated should include, as a minimum, stall, runaway, lockup, interruption/restoration, clock and timing faults, counter overflow, missing/corrupt data, and effects of hardware faults on software.(15B4)

430.58

The accident analyzed under this section considers only the airborne radioactivity that may be released due to potential failure of a concentrated waste tank in the radwaste enclosure. The SRP acceptance criteria, however, requires demonstration that the liquid radwaste concentration at the nearest potable water supply in an unrestricted area resulting from transport of the liquid radwaste to the unrestricted area does not exceed the radionuclide concentration limits specified in 10 CFR Part 20, Appendix B Table II, Column 2. Such a demonstration will require information on possible dilution and/or decay during transit which, in turn, will depend upon site specific data such as surface and ground water hydrology and the parameters governing liquid waste movement through the soil. Additionally, special design features (e.g.,

steel liners or walls in the radwaste enclosure) may be provided as part of the liquid radwaste treatment systems at certain sites. The staff will, therefore, review the site specific characteristics mentioned above individually for each plant referencing the ABWR and confine its review of ABWR, only to the choice of the liquid radwaste tank. Therefore, provide information on the following: (15.7.3)

(a) Basis for determining the concentrated waste tank as the worst tank (this may very well be the case, but in the absence of information on the capacities of major tanks, particularly the waste holdup tanks, it is hard to conclude that the above tank both in terms of radionuclide concentrations and inventories will turn out to be the worst tank).

(b) Radionuclide source terms, particularly for the long-lived radionuclides such as Cs-137 and Sr-90 (these may be the critical isotopes for sites that can claim only decay credit during transit) in the major liquid radwaste tanks.

440.108

Provide further justification for the fact that the input parameters and initial conditions for analyzed events are conservative. Provide a list of what parameters will be checked at startup and which will be in the Technical Specifications. You should define the range of operating conditions and fuel types for which your input parameters will remain valid. For example, would these parameters valid for 9x9 or 7x7 fuel or similar large change in the fuel lattice. (15)

470.1

Subsection 15.6.2 of the ABWR FSAR provides your analysis for the radiological consequences of a failure of small lines carrying primary coolant outside of containment. This analysis only considers the failure of an instrument line with a 1/4-inch flow restricting orifice. Show that this failure scenario provides the most severe radioactive releases of any postulated failure of a small line. Your evaluation should include lines that meet GDC 55 as well as small lines exempt from GDC 55.

470.2

Provide a justification for your assumption that the plant continues to operate (and therefore no iodine peaking is experienced) during a small line break outside containment (Subsection 15.6.2) accident scenario. Also provide the basis for the assumption that the release duration is only two hours.

470.3

Subsection 15.6.4.5.1.1 of the FSAR gives the iodine source term (concentration and isotopic mix) used to analyze the steam line break outside of containment accident. The noble gas source term, however, is not addressed. Provide the noble gas source term used. Also, the table in Subsection 15.6.4.5.1.1 seems heavily weighted to the shorter lived activities (i.e., (I-134). Provide the bases for the isotopic mix used in your analysis (iodine and noble gas).

470.4

Subsection 15.6.5.5 states that the analysis is based on assumptions provided in Regulatory Guide 1.3 except where noted. For all assumptions (e.g., release assumed to occur one hour after accident initiation, the chemical species fractions for iodine, the temporal decrease in primary containment leakage rates, credit for condenser leakage rates, and dose conversion factors) which deviate from NRC guidance such as regulatory guides and ICRP2, provide a detailed description of the justification for the deviation or a reference to another section of the SSAR where the deviations are discussed in detail. Provide a comparison of the dose estimates using these assumptions versus those which would result from using the NRC guidance.

470.5

Provide a discussion of, or reference to, the analysis of the radiological consequences of leakage from engineered safety feature components after a design basis LOCA.

470.6

For the spent fuel cask drop accident, what is the assumed period for decay from the stated power condition? What is the justification for that assumption?

470.7

The tables in Chapter 15 should be checked and revised as appropriate. In several cases the footnotes contain typographical errors related to defining the scientific notation. Table 15.7-12 also appears to contain inappropriate references to Table 15.7-16, rather that Table 15.7-13.

470.8

It is stated that Regulatory Guides 1.3 and 1.45 were used in the calculations of X/Q values. Based on the values presented, it appears as though a Pasquill stability Class F and one meter per second wind speed were assumed, with adjustment for meander per Figure 3 of Regulatory Guide 1.145. If this is not the case, describe the assumptions and justification used in calculating the X/Q values which are used in Chapter 15 dose assessments.

470.9

The SGTS filter efficiencies of 99% for inorganic and organic iodine are higher than the 90% and 70% values, respectively, assumed in Regulatory Guide 1.25 if it can be shown that the building atmosphere is exhausted through adsorbers designed to remove iodine. Provide a justification for the use of the higher values.

470.10

Dose related factors such as breathing rates, iodine conversion factors and finite versus infinite cloud assumptions for calculating the whole body dose are not stated explicitly, although reference is made to Regulatory Guide 1.25 and another document. State these assumptions explicitly and justify use of any values which deviate from Regulatory Guide 1.25.

440.109

Provide an analysis of the loss of instrument air (nitrogen). (15)

440.110

In SSAR Table 15.0-2, the following transients are not categorized as moderate frequency event [Category (a)]

(a) Runout of two feedwater pumps (Cat.c)

(b) Opening of all Control and Bypass Valves (Cat.c)

(c) Pressure Regulator Downscale failure (Cat.c)

(d) Generator Load Rejection, Failure of One Bypass Valve (Cat.b)

(e) Generator Load Rejection with Bypass Off (Cat.c)

(f) Turbine Trip with Failure of One Bypass Valve (Cat.b)

(g) Turbine trip, Bypass Off (Cat.c)

(h) Loss of Aux. Power Transformer and one S/up transformer (Cat c)

(i) Trip of all Reactor Internal Pumps (Cat.c)

(j) Fast Runback of all Reactor Internal Pumps (Cat.c)

(k) Inadvertent HPCF pump start-up (Cat.b)

Category b refers to Infrequent event and Category c refers to limiting faults.

The above categorization of transients is a significant deviation from the SRP and hence sufficient justification must be submitted to support the change in the categorization. (15)

440.111

Provide a table similar to 15.0-2 showing your evaluation of anticipated transients with single failure. List the single failure chosen for each event and provide a justification for why the chosen failure is the most limiting. (15)

440.112

Provide the following:

(1) A listing of all equipment which is not classified as safety-related but is assumed in FSAR analyses to mitigate the consequences of transients or accidents.

(2) Justification for the assumption of operability of this equipment based upon equipment quality, reliability, and proposed surveillance requirements.

(3) Discuss the consequences of those events concerning (i) number of fuel failures, (ii) delta CPR and (iii) delta perk pressure that would result if only safety grade systems or components were considered in the specific transients analyses taking credit of non-safety grade systems or components. (15)

440.113

You have classified the trip of all reactor internal pumps as a limiting fault. This is based on your assumption that the loss of greater than three reactor internal pumps is 10^{-6} per year. Provide operating experience data to justify this failure rate. (15)

440.114

The ABWR feedwater control system and the steam bypass and pressure control system use a triplicated digital system. You claim that no single failure in these systems will cause a minimum demand to all turbine control valves and bypass valves or the runout of two feedwater pumps. (15)

(a) What is the reliability of the system?

(b) What design feature of these systems prevent common mode failure to more than one channel?

(c) What protection is provided in these systems against a technician disabling a second channel while performing maintenance on the first.

(d) What are the most limiting events for the case where two channels are lost in these systems?

440.115

Provide further analysis and numerical justification for your assertion that FMCRD design is equivalent to an ARI system and that the SLCS is not required to respond to an ATWS. (15)

440.116

For each transient and accident, identify the computer code used in the analysis in the respective section of Chapter 15. (15)

470.5

Provide a discussion of, or reference to, the analysis of the radiological consequences of leakage from engineered safety feature components after a design basis LOCA.

470.6

For the spent fuel cask drop accident, what is the assumed period for decay from the stated power condition? What is the justification for that assumption?

470.7

The tables in Chapter 15 should be checked and revised as appropriate. In several cases the footnotes contain typographical errors related to defining the scientific notation. Table 15.7-12 also appears to contain inappropriate references to Table 15.7-16, rather that Table 15.7-13.

470.8

It is stated that Regulatory Guides 1.3 and 1.45 were used in the calculations of X/Q values. Based on the values presented, it appears as though a Pasquill stability Class F and one meter per second wind speed were assumed, with adjustment for meander per Figure 3 of Regulatory Guide 1.145. If this is not the case, describe the assumptions and justification used in calculating the X/Q values which are used in Chapter 15 dose assessments.

470.9

The SGTS filter efficiencies of 99% for inorganic and organic iodine are higher than the 90% and 70% values, respectively, assumed in Regulatory Guide 1.25 if it can be shown that the building atmosphere is exhausted through adsorbers designed to remove iodine. Provide a justification for the use of the higher values.

470.10

Dose related factors such as breathing rates, iodine conversion factors and finite versus infinite cloud assumptions for calculating the whole body dose are not stated explicitly, although reference is made to Regulatory Guide 1.25 and another document. State these assumptions explicitly and justify use of any values which deviate from Regulatory Guide 1.25.

440.109

Provide an analysis of the loss of instrument air (nitrogen). (15)

440.110

In SSAR Table 15.0-2, the following transients are not categorized as moderate frequency event [Category (a)]

(a) Runout of two feedwater pumps (Cat.c)

(b) Opening of all Control and Bypass Valves (Cat.c)

(c) Pressure Regulator Downscale failure (Cat.c)

(d) Generator Load Rejection, Failure of One Bypass Valve (Cat.b)

(e) Generator Load Rejection with Bypass Off (Cat.c)

(f) Turbine Trip with Failure of One Bypass Valve (Cat.b)

(g) Turbine trip, Bypass Off (Cat.c)

(h) Loss of Aux. Power Transformer and one S/up transformer (Cat c)

(i) Trip of all Reactor Internal Pumps (Cat.c)

(j) Fast Runback of all Reactor Internal Pumps (Cat.c)

(k) Inadvertent HPCF pump start-up (Cat.b)

Category b refers to Infrequent event and Category c refers to limiting faults.

The above categorization of transients is a significant deviation from the SRP and hence sufficient justification must be submitted to support the change in the categorization. (15)

440.111

Provide a table similar to 15.0-2 showing your evaluation of anticipated transients with single failure. List the single failure chosen for each event and provide a justification for why the chosen failure is the most limiting. (15)

440.112

Provide the following:

(1) A listing of all equipment which is not classified as safety-related but is assumed in FSAR analyses to mitigate the consequences of transients or accidents.

(2) Justification for the assumption of operability of this equipment based upon equipment quality, reliability, and proposed surveillance requirements.

(3) Discuss the consequences of those events concerning (i) number of fuel failures, (ii) delta CPR and (iii) delta perk pressure that would result if only safety grade systems or components were considered in the specific transients analyses taking credit of non-safety grade systems or components. (15)

440.113

You have classified the trip of all reactor internal pumps as a limiting fault. This is based on your assumption that the loss of greater than three reactor internal pumps is 10^{-6} per year. Provide operating experience data to justify this failure rate. (15)

440.114

The ABWR feedwater control system and the steam bypass and pressure control system use a triplicated digital system. You claim that no single failure in these systems will cause a minimum demand to all turbine control valves and bypass valves or the runout of two feedwater pumps. (15)

(a) What is the reliability of the system?

(b) What design feature of these systems prevent common mode failure to more than one channel?

(c) What protection is provided in these systems against a technician disabling a second channel while performing maintenance on the first.

(d) What are the most limiting events for the case where two channels are lost in these systems?

440.115

Provide further analysis and numerical justification for your assertion that FMCRD design is equivalent to an ARI system and that the SLCS is not required to respond to an ATWS. (15)

440.116

For each transient and accident, identify the computer code used in the analysis in the respective section of Chapter 15. (15)

20.2.16 Chapter 16 Questions

None.

20.2.17 Chapter 17 Questions

260.1

General Electric's commitment to QA-related Regulatory Guides (RGs) is given in Table 17.0-1. In accordance with Chapter 17 of the Standard Review Plan (NUREG-0800), a commitment to RGs 1.8, 1.26, and 1.29 should also be made. This can be done by referencing another section of the Safety Analysis Report.

260.2

Clarify why Table 17.0-1 shows RG 1.94 as not applicable to the ABWR scope while similar RGs for installation, inspection, and testing (RGs 1.30 and 1.116) are shown as being applicable.

260.3

Revision 3 of RG 1.28 states, “Applicants and licensees may commit to follow either the ANSI/ASME N45.2-series standards or the ANSI/ASME NQA-1-1983 standard, but not a combination of the two.” Table 17.0-1 indicated GE's commitment to the N45.2-series standards, but the third paragraph of SSAR Section 17.0 states that the terms and definitions of NQA-1 apply and SSAR section 17.1 refers throughout to NQA-1. Clarify weather GE is commits to the N45.2-series standards referenced in RG 1.28 (As clarified in Reference 1), to the NQA-1 standard, or to both.

20.2.18 Chapter 18 Questions

620.1

Describe GE's human factors design team, the staff's human factors expertise, and its responsibilities for human factors on the ABWR design.

620.2

Both Hitachi and Toshiba are designing main control room workstations which, although based upon the "common engineering studies, may result in two different workstation design implementations within one two-unit control room. Describe the process that GE will use to actually implement high-level, single-unit workstation requirements and design selection, including the decision process to be followed in selecting the Hitachi or Toshiba approach, a hybrid, or a different design.

620.3

Describe how the GE/US ABWR differs from the Japanese versions in terms of the human factors operations considerations. For example, it is our understanding that the Japanese do not use symptom-based procedures which are essential to accident management in U.S. plants; this difference will presumably have an influence on workstation design.

620.4

The control room will make use of many advanced hardware and software technologies for which the nuclear industry has little experience. Describe the process that GE will use to demonstrate that these technologies are being properly used and will not adversely effect human performance.

620.5

The EPRI ALWR requirements document and several of the GE documents provided during the March 6-7, 1990 meeting speak about optimizing operator performance. Describe how operator performance is defined in terms of performance parameters and the measures to be used to quantify these parameters. Describe how this information will be factored into the design process in a timely fashion.

620.6

It appears that the workstation design may precede procedure design (which has historically been the case in the nuclear industry). Yet, it seems that GE has the opportunity to follow the potentially valuable path of specifying what the operator has to accomplish in the control room to a great level of detail (via detailed task analyses and implementing procedures) and then design a workstation that will best support those operator tasks. Describe the temporal relationship between the future development of the operating procedures and the design of the workstation.

620.7

The PRA can provide insights about the most significant human errors in terms of their effects on plant safety. With this knowledge the human-system interface can be designed to help mitigate the effect of the errors and to make the system more tolerant to errors which have occurred. Describe how the results and in sights derived from the PRA are being used to support the control room design.

620.8

Describe the content and format of training materials to be provided by GE to purchasers of the ABWR. Will these materials be offered as customized options, or will they be included and standardized?

620.9

Describe the role of GE in the development of normal, abnormal and emergency operating procedures, including the generic technical basis document and writers guide, the development of procedures generation documents, the verification and validation process, and the procedures maintenance program. Will GE develop sample procedures or offer a package of procedures to be modified based on site-specific technical considerations?

620.10

Describe how the analysis of functions will determine a proper balance of automated and manual tasks to ensure an appropriate operator work load.

620.11

Describe the decision criteria used to select tasks for analysis, and describe how the task analyses were organized.

620.12

Describe the criteria used for the selection of specific accident scenarios/sequences for which task analyses were performed and identify the scenarios/sequences which were analyzed.

620.13

Very detailed procedures for function and task analyses were developed by the ABWR team (Ref PPE ITEM NO. 5.1.3). However, the task analysis report provided for the Nuclear Boiler System (Ref. PPE Item No. 3.9B, Rev. 0, 9/22/89) supplied considerably less detail than that specified in the procedure. The analysis report specifies that it was conducted in accordance with a list of reference documents: but this list does not include the procedures document. While the report does identify monitoring and control requirements and makes recommendations for automation, it does not provide timelines or workload estimates needed from the task analysis for other design and analysis activities, such as the HRA. Discuss why the detailed task analysis procedure was not followed and the consequences of this decision.

620.14

Discuss the technical basis for single-operator operations with regard to the requirements of 10 CFR 50.54(m), and the following issues:

(a) The control room technology developments which would enable this approach;

(b) The analyses that will be performed to assure that safety will not be compromised.

620.15

Describe how the plant addresses the single-failure criterion with a single operator.

620.16

Which existing BWR is most similar to the ABWR with regard to the role of the operations staff? Discuss any significant differences that exist between ABWR operations and operations at this most similar existing BWR.

620.17

Describe the implications for operator selection and training based upon the ABWR's use of increased automation, advanced instrumentation and control and compact workstations.

620.18

With increases in automation in complex systems which change the operator's role from that of an active "in-the-loop" controller to that of a systems monitor, human factors practitioners have frequently identified new problems, including:

(a) Maintaining an appropriate level of work load;

(b) Maintaining vigilance in system monitoring;

(c) Maintaining adequate awareness of system status so that the operator ran intervene and take over system operation when required;

(d) Maintaining specialized skills.

Discuss how each of the above issues will be addressed.

620.19

While the plant is under automated control and an abnormal condition such as a reactor scram occurs, the Power Generation Control System (PGCS) alerts the operator and drops out of automated mode. Describe the time period over which this change occurs. Since the PGCS controls many systems, describe the implications for operator workload, and subsequent to, the time of the status change.

620.20

The major driving force affecting control room design appears to be the concept of one-person operations during normal conditions. This leads to the requirement to consolidate most of the monitoring and control capability into a single, relatively compact work station in contrast to the traditional analog control boards. This approach then leads to requirements to minimize dedicated controls and displays (because of limited real estate at the work station), utilize soft controls (to replace dedicated controls), utilize CRT-like display devices which only display a limited set of plant data at a time (to replace instrument displays) and to utilize intelligent operator aids based upon expert systems, etc. to assist the one operator to accomplish his tasks. While these technologies may have merits of their own, we are concerned about the appropriateness of this technology as a design driver for U.S. plants. Please discuss your rationale for this concept.

620.21

One of the main features of the control room is the use of a computer-based work station in place of the traditional control boards with dedicated controls and displays. With such an approach, the methods by which information is displayed to the operator via CRTs and other display devices is of critical importance. Indeed, the display of information and the methods by which the operator interacts with that information are arguably the most important aspects of the control room design. Yet, most of the information presented by GE thus far concerning control room and work station design has emphasized the hardware, ergonomics and anthropometrics of the design. Little information has been made available on the display design and human-software interface. Much more information is needed in order to evaluate the adequacy of the control room to support the operator’s tasks. Please describe the approach that you will use to determine th.e following:

(a) The planning and control of the interaction between the operator and system information

(b) The design basis for the interface (e.g., command language or direct manipulation)

(c) Planning and design of high-level data integration

(d) Operator access to information and the parameters that will be optimized in the design of the interface (e.g., speed of data access)

(e) Any data that will not be accessible to operators

(f) Display techniques for various types of data

(g) Coding methods to be used

620.22

Describe how the requirements for: (1) information/data display and (2) methods by which the operator will interact with the system will be reflected in hardware design requirements. It

appeared from the material presented by GE on March 6-7, 1990, that hardware requirements were preceding these issues.

620.23

With regard to the design of the control room:

(a) Was a human factors design guideline developed specifically for the design of the human-software/information interface, as discussed in Question 620.21, above?

(b) Was a human factors design guideline developed specifically for the ABWR to assist in control of the interface design, or were the ABWR human factors design guidelines derived from human factors design guidelines available in the literature? If neither, how were the ABWR guidelines developed? If existing guidelines were used, please identify them and provide the audit trail.

(c) How were guidelines developed for those interface characteristics for which there appear to be no existing guidelines in the literature?

620.24

A significant feature of the ABWR control room design is the use of advanced and intelligent operator aids based upon expert systems and other AI technologies. With respect to these operator aids, please describe the following:

(a) The extent of the dependence on intelligent operator aids that is necessary to achieve the single-operator design goal

b) The specific operator aids that are planned and the technology on which they are based

(c) The methods of knowledge engineering that will be used and the steps that will be taken to assure that all appropriate knowledge will be incorporated into the database

(d) The approach to be taken to develop operator confidence in the systems to assure that they will be appropriately utilized

(e) The approach to be taken to minimize undue reliance on and blind acceptance of these systems

(f) The methods to be used for the verification and validation of the performance of intelligent operator aids

620.25

The workstation will have a few dedicated controls and displays (C/Ds). Describe the rationale and analyses being used to determine which C/Ds will be dedicated and which will be “soft”.

620.26

Computer-based work stations can often present data interface management problems to the operator (such as the operator spending too much time managing data windows rather than monitoring plant information) which reflect a shift from task-related workload to interface-management workload. Describe how the design of the work station controls and displays will minimize the workload associated with the operator's management of the interface. Discuss any assistance that the operator will have in calling up the appropriate displays via automatic display "triggers" or an expert system.

620.27

It appears that alarm information is being presented in three separate locations: on the large display screen, on dedicated alarms and on CRTs. With respect to annunciator warning systems data, please discuss:

(a) How allocation of alarm information to the above locations is determined and which alarms are located where

(b) How the CRT-based alarms will be presented

(c) How alarm information will be prioritized

(d) Whether alarm filtering will be used and, if so, (1) by what methods, and (2) whether operators will have access to filtered-out alarm data

620.28

Describe any trade studies and/or investigations which have been performed to support the selection of the approaches to display and control being planned for the control room, including, for example, the use of touch panel control for specific functions.

620.29

Describe how data protection and security will be assured.

620.30

The control room will have only a single command workstation. Discuss why there is no back-up as recommended in the EPRI ALWR Chapter 10 requirements document. In addition, please discuss the following:

(a) Any loss of monitoring and control functions that have been analyzed and their initiating events

(b) Whether any single event could cause the loss of a major portion of the workstation and/or the loss of monitoring and control functions

(c) The effects of the loss of one or two CRTs at the workstation including whether this could require too much information to be displayed at the remaining display devices

(d Whether awkward control/display relationships and awkward operations could result from the loss of any small section of the workstation

620.31

Since there is only one workstation, and it is typically manned by a single operator, describe any analysis that have been performed to assure that the workstation can appropriately accommodate two-person operations during accident scenarios. Please include the following in the discussion:

(a) How the responsibilities and tasks are laid out to assure well-coordinated two-person operations

(b) Any function or task analyses that have been performed to assure that the two operators will not have unintended and unwanted interactions

(c) How emergency operating procedures (EOPs) will account for one and two-person operations

620.32

Although an advanced computer-based control room is planned, the design of the remote shutdown panels will be based upon conventional hardware (e. g., hard control devices, analog indicators, etc.). Based upon the March 6-7, 1990 presentation by G. E., it appears that this diversity was a design goal. Discuss the technical basis for this approach, including the human factors implications such as:

(a) Likely confusion due to the differences between operations in the control room and at the RSP;

(b) Increased training burden and operator burden associated with the need to learn two different systems, one of which will be used constantly and the other very infrequently, if ever.

620.33

Describe the design of the other local control panels, given the dual approach discussed above.

620.34

Discuss the technical basis for the design of local valve operations, including the determination of local vs. control room position indications.

620.35

Discuss how TMC operations are changed in the design of the ABWR when compared with a standard BWR.

620.36

Discuss the criteria used to determine which instrumentation will be manually calibrated.

620.37

Discuss the criteria used for the selection of computerized test operations.

20.2.19 Chapter 19 Questions

210.51

The information in this section should be revised to more nearly reflect the current status of this issue. GSI II.E.6.1 originally consisted of the following sub-issues:

(1) In-situ testing of motor operated valves (MOV)

(2) In-situ testing of pressure isolation valves (PIV)

(3) Reevaluation of thermal overload protection devices for motor operated valves.

(4) In-situ testing of check valves

Sub-issues 1, 2 and 3 are no longer considered to be part of II.E.6.I. Sub-issue I was subsumed by the staff's evaluation of responses to Generic Letter 89-10, "Safety-Related MOV Testing and Surveillance". Sub-issue 2 was subsumed by Generic Safety Issue 105, "Interfacing Systems LOCA in Light-Water Reactors". Sub-issue 3 is considered to be resolved for the ABWR on the basis of the unconditional commitment in the SSAR Table 1.8-20 to Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor-Operated Valves". Sub-issue 4 remains unresolved at this time. During a meeting on April 7, 1986 between the staff and industry representatives, it was agreed that industry would initiate an aggressive program to resolve the check valve issue. Since that time, the Institute (EPRI), the Nuclear Power Operation (INPO), the Electric Power Research Institute (PERI), the Nuclear Industry Check Valve Group (NIC) and the staff have made some progress in addressing this issue. However, as stated in a letter to Mr. Z. T. Pate, President of INPO, dated April 20, 1990, the staff continues to find weakness in the efforts of individual licensees to improve the performance of check valves. To assist the staff in its continuing evaluations and perspectives regarding the resolution of the check valve issue. The staff has not yet received a complete response to this request.

The staff does not agree that the information in the "ABWR Resolution" of Subsection 19B.2.2 in the SSAR is sufficient to resolve this issue for the ABWR. The exceptions to position indication of check valves will require some clarification. However, the staff prefers that this type of information be included as a part of the ASME Section XI Inservice Test Program for safety-related pumps and valves which is discussed in the SSAR, Subsection 3.9.6. Therefore, GE is requested to revise Subsection 19B.2.2 related to sub-issue 4 to reflect a more broad commitment to the collective industry and NRC activities relative to implementation of the resolution of issues on in-situ testing of check valves. In addition, the staff will need to complete its review of the ABWR Inservice Testing Program before this issue can be considered resolved.

Since sub-issue 1 has been submitted, Subsection 19B.2.2 should also include a commitment to provide a response to Generic Letter 89-10 which will be applicable to the ABWR. (19B.2.2)

210.52

Recent BWR operating experience indicates that the isolation valves between the RCS and low pressure interfacing systems may not adequately protect against overpressurization of low pressure systems.

For ABWRs, pressure isolation valve instrumentation and controls are provided to (1) prevent opening shutdown cooling connections to the vessel in any loop when the pool suction valve, discharge valve, or spray valves are open in the same loop, (2) prevent opening the shutdown connections to and from the vessel whenever the RCS pressure is above the shutdown range, (3) automatically close shutdown connections when RCS pressure rises above the shutdown range, and (4) prevent operation of shutdown suction valves in the event of a signal that the water level in the reactor is low.

The ABWR has been designed to minimize the possibility of an interfacing system LOCA in the following ways. The low pressure systems directly interfacing with the RCS are designed with 500 psig piping which provides for a rupture pressure of approximately 100 psig. In addition, the high/low-pressure motor-operated isolation valves have safety-grade, redundant pressure interlocks. Also, the motor-operated emergency core cooling system (ECCS) valves will only be tested when the reactor is at low pressure. All inboard check valves on the ECCS will be testable and have position indication. Additionally, design criteria used by GE require that all pipe designed to 1/3 or greater of reactor pressure requires two malfunctions to occur before the pipe would be subjected to reactor system pressure. The pipe designed to less than 1/3 reactor pressure requires at least three malfunctions before the pipe would be subjected to reactor system pressure.

Position—Since ABWR low pressure systems are designed only for 500 psig rather than the full RCS design pressure of 1250 psig, the ABWR design should provide (1) the capability for leak testing of the pressure isolation valves, (2) valve position indication that is available in the control room when isolation valve operators are deenergized and (3) high-pressure alarms to warn control room operators when rising RCS pressure approaches the design pressure of attached low-pressure systems and both isolation valves are not closed. It is the staff's position that GE should confirm that the above design features are incorporated into the ABWR design.

GI-96 was related to PWRs which considers the failure of the low pressure isolation valves between the RCS and RHR system in PWRS. The issues contained in GI-96 now are incorporated into GI-105. (19B.2.15)

210.53

Position—The record of relief-valve failures to close for all boiling-water reactors (BWRs) in the past 3 years of plant operation is approximately 30 in 73 reactor-years (0.41 failures per reactor-year). This has demonstrated that the failure of a relief valve to close would be the most likely cause of a small-break loss-of-coolant accident LOCA). The high failure rate is the result of a high relief-valve challenge rate and a relatively high failure rate per challenge (0.16 failures per challenge). Typically, five valves are challenged in each event. This results in an equivalent

failure rate per challenge of 0.03. The challenge and failure rates can be reduced in the following ways:

(1) Additional anticipatory scram on loss of feedwater,

(2) Revised relief-valve actuation setpoints,

(3) Increased emergency core cooling (ECC) flow,

(4) Lower operating pressures,

(5) Earlier initiation of ECC systems,

(6) Heat removal through emergency condensers,

(7) Offset valve setpoints to open fewer valves per challenge,

(8) Installation of additional relief valves with a block- or isolation-valve feature to eliminate opening of the safety/relief valves (SRVs), consistent with the ASME Code,

(9) Increasing the high steam line flow setpoint for the main steam line isolation valve (MISV) closure,

(10) Lowering the pressure setpoint for MSIV closure,

(11) Reducing the testing frequency of the MSIVs,

(12) More-stringent valve leakage criteria, and

(13) Early removal of leaking valves.

An investigation of the feasibility of reducing challenges to the relief valves by use of the aforementioned methods should be conducted. Other methods should also be included in the feasibility study. Those changes which are shown to reduce relief-valve challenges without compromising the performance of the relief valves or other systems should be implemented. Challenges to the relief valves should be reduced substantially (by an order of magnitude).

220.16

Generic safety issue 82 “Beyond Design Basis Accidents in Spent Fuel Pools” is concerned with the loss of the pool water which may result in a fire in the pool causing a release of fission products. In the ABWR resolution, it is indicated that the spent fuel pool will be designed to withstand a design basis earthquake without pool drainage, and will be arranged to prevent cask movement over the pool, which will be accomplished through the use of a separate cask loading pit. Was a cask drop in the cask loading pit considered? Since the cask loading pit is adjacent to the spent fuel pool. In addition, it appears that the fuel pool is near the staging area for the

reactor vessel head, indicate the effect on the fuel pool of vessel head drop on the adjacent staging area. (19B.2.14)

220.17

Generic Safety Issue No. 103 “Design for Probable Maximum Precipitation” (PMP) is concerned with the difference in the determination of PMP. BY using the recently developed NOAA/NWS procedures which are believed to be more realistic, PMP estimates larger than those obtained by previously used methodologies may lead to higher flood levels. Therefore, in ABWR resolution on Page 19B 2-47, specify that the recently developed NOAA/NWS procedures will be used for determining PMP for a specific site. (19B.2.17)

252.16

(1) The applicant should define bolting in detail. Bolting in this context should include bolts, studs, embedments, machine/cap screws, threaded fasteners, and associated nuts and washers.

(2) Define high strength bolting and medium strength bolting in terms of material and mechanical properties.

(3) Provide bolting manufacture process (e.g., heat treated, quenched, tempered, etc.).

(4) Provide bolting manufacture process (e.g., equipment and piping systems) where the high strength bolting or medium bolting will be used.

(5) Discuss how to avoid the intergranular stress corrosion cracking (IGSCC) of bolting in a BWR hydrogen environment.

(6) Identify thread lubricants that will be used and identify chemical compound(s) in them.

(7) The applicant discussed the ALWR Resolution initiated by the Atomic Industrial Forum/Metal Properties Council Task Group and BWR Requirements in the EPRI-ALWR Requirements Document. It is unclear whether the applicant will follow the resolutions and requirements. (19B.2.12)

260.4

The ALWR Resolution Summary for issues I.F.1 and II.F.5 states:

(1) The designer shall identify any structures, systems, or components (items) that are not safety related but for which provisions beyond normal industry practice are judged to be needed to provide desired reliability and availability.

(2) At the same time, specific surveillance, maintenance provisions (appropriate for specific item and desired reliability and availability) shall be identified for those items.

The NRC evaluation is that ALWRs should have a Reliability Program to ensure that the facility is operated and maintained within enveloping PRA assumptions throughout its life. The NRC anticipates that these new (Reliability Program) requirements will effectively subsume the I.F.1 and II.F.5 issues and these issues can be considered resolved.

The ABWR Resolution states:

(1) The ABWR application of quality system requirements satisfies the ALWR resolution.

(2) An interface requirement (Section 19B.3.1) is included to ensure that quality system requirements will be provided during construction and operation.

(3) Therefore, this issue is resolved for the ABWR.

Request for Additional Information 1

It is not clear to the staff that the ABWR SSAR describes how points 1 and 2 of the ALWR Resolution Summary (above) are to be satisfied. That is, how is the ABWR designer identifying items for which provisions beyond normal industry practice are judged to be needed? And how are specific surveillance/maintenance provisions being identified for those items? SSAR Table 3.2-1 is used to show the quality assurance that is applied to plant items. The table indicates that a quality assurance program meeting 10CFR50 Appendix B either does or does not apply. In some instances, where Appendix B does not apply, there is reference to a footnote regarding quality assurance. Such references are neither wide-spread enough nor specific enough to really meet an objective of the classification system which is to assign appropriate Quality Control and Quality Assurance measures.

The SSAR should be clarified in this regard, or justification should be given for not doing so. For example, footnote “u” regarding quality assurance for non-safety-related fire protection items should make it clear that a quality assurance program meeting the guidance of Branch Technical Position CMEB 9.5-1 (NUREG-0800) will be applied to each such item. Similarly, for non-safety-related radioactive waste management items, a footnote should make it clear that a quality assurance program meeting the guidance of Regulatory Guide 1.143 will be applied during design and construction. The safety parameter display system (or its equivalent), though not safety-related, should have a quality assurance program beyond normal industry practice applied, and this should be clear in Table 3.2-1. Generic Letter 85-06, “Quality Assurance Guidance for ATWS Equipment That Is Not Safety Related,” is also applicable. If GE has not already done so, it should ascertain whether there are other ABWR plant items within the scope of points 1 and 2 of the ALWR Resolution summary (above) and revise Table 3.2-1 accordingly

if required. Then the ABWR Resolution should reference Table 3.2-1 to show how GE has resolved TMI issues I.F.I. and II.F.5 for the ABWR.(19B.2.1)

260.5

The statement in the ABWR SSAR, “Applicants referencing the ABWR design shall have a Quality Assurance Program satisfying the requirements of Section 19.B.2.1(2) including the right to impose additional environmental requirements,” does not appear to accurately reflect the requirements of 19B.2.1(2). The “right to impose additional environmental requirements” is not as encompassing as “the right to impose additional requirements to supplement the 10 CFR 50, Appendix B requirements.” It is not clear what is meant by “environmental” requirements.

Request for Additional Information 2

Clarify that applicants referencing the ABWR design shall have the right to impose additional requirements to supplement the 10 CFR 50, Appendix B requirements, or justify not doing so. (19B.3.1)

260.6

The response to this item states: “Interface requirement, see Subsection 19A.3.6.” Subsection 19A.3.6 states: “. . . (Reference Subsection 19A.2.4.)” The response to Subsection 19A.2.4 states: “. . . This requirement is not applicable to the ABWR. It applies only to PWR-type reactors.” This series of references takes us from an NRC requirement that procedures provide for evaluation and feedback of related experience in a timely manner to the ABWR designers and constructors to a conclusion that there is no requirement.

Request for Additional Information 3

Clarify the response to 19.2.41, paying particular attention to the references, or justify not doing so. (19A.2.41)

260.7

The response states: “This issue is addressed in Appendix 19B.”

Request for Additional Information 4

Since Appendix 19B has many pages, please narrow the reference to item 19B.2.1 in Appendix 19B. (19A.2.42)

420.137

Position—Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement existing instrumentation (including primary coolant saturation monitors) in order to provide an unambiguous, easy-to-interpret indication of inadequate core cooling (ICC). A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.

Resolution—The staff requires the following additional information to complete the review on this item:

GE refers to Topical Report SLI-8211, “Review of BWR Reactor Vessel Water Level Measurement System,” in their response. Generic Letter No. 84-23, “Reactor Vessel Water Level Instrumentation for BWRs,” discusses in detail the potential improvements required as a result of the topical report review. How the ABWR design satisfies the requirements of GL No. 84-23 should be explained in detail. (19A.2.30)

430.300
Justify non-compliance of the current ABWR design with position 7 of TMI issue II.E.4.2. This design does not include the containment isolation on a high containment radiation signal for the containment purge and vent isolation valves as required by position 7. (19A.2.26, 19A.2.27)

430.301
Explain whether the reopening of isolation valves is performed on a valve-by valve basis (which is acceptable according to the guidance provided by TMI issue II.E.4.2) or as a ganged opening (which is not acceptable). (19A.2.26, 19A.2.27)

430.302
Discuss the administrative controls that will be in effect to assure that closed purge isolation valves cannot be inadvertently opened. (19A.2.26, 19A.2.27)

430.303
Explain the technical basis for the 72 (versus 24) hour technical specification limit allowing the large diameter purge lines to be open above 15% power at the beginning and end of the fuel cycle. Since these valves can be open during power operation, verify that the large diameter (22") purge line isolation valves can successfully perform their intended function under accident conditions (containment design pressures). (19A.2.26,19A.2.27)

430.304
Evaluate the adequacy of two-inch at-power purge lines for relieving primary containment excessive pressure (resulting from a combination of compressed air system leaks, steam leaks, and elevated containment atmosphere temperatures associated with hot days or degraded containment HVAC performance) during normal operation in light of current reactor operating experience in which many plant operators are forced to periodically open the large containment purge lines at power to maintain normal containment pressure. Include an evaluation of the ability of the two-inch lines to maintain normal containment pressure, the possible need to operate the large (22") containment purge and vent isolation valves, and identify the size of a small purge line required to preclude operation of the 22" containment purge lines during power operation. (19A.2.26,19A.2.27)

430.305

Also note a discrepancy between Figure 6.2-39a sheet 1 and Table 6.2-7 of Amendment 11. Valve T31-F007 is listed as a 2" valve in Table 6.2-7. However, Figure 6.2-39a has been modified and this valve now appears to be on the same line “as the 14" rupture disk to be used for containment overpressure protection system. Other information provided for this valve in the figure and the table is also contradictory. Modify the figure and table to accurately represent both the 2" at-power purge lines and the containment overpressurization protection flow path. (19A.2.26, 19A.2.27)

430.306

Widely separated primary containment penetrations for the drywell and wetwell purge systems (supply side penetrations X-80 and X-240; and exhaust side penetrations X-81 and X-241, SSAR Figure 6.2-39a, (sheet 1) have common primary containment outboard isolation valves. Explain how the above configurations comply with GDC 56 which requires the outboard isolation valve to be as close to the containment (i.e., the drywell or wetwell in this case) as practical. (19A.2.26, 19A.2.27)

430.307

Section 19B.2.6 of the ABWR SSAR reflects the EPRI Requirements Document positions on hydrogen generation, that is, containment concentrations resulting from an active fuel-clad oxidation of 75% and the concentration of less than 13%. These requirements are less conservative than the requirements of 10 CFR 50.34(f), namely 100% and 10%. Also, provide an analysis and supporting documentation demonstrating that the hydrogen control system will be able to maintain containment atmosphere within acceptable limits and that the hydrogen recombiners will function in an extremely hydrogen rich environment, using the hydrogen generation rates and allowable concentrations of 10 CFR 50.34(f). (19A.2.12, 19A.2.21, 19A.2.46, 19B.2.6)

430.308

There is a discrepancy between ABWR SSAR Section 19A.2.12 which states that permanently installed recombiners are provided and Section 6.2.1.1.1 which states that portable recombiners will be available for use after a LOCA signal is generated (Section 6.2.5.2.7 also states that recombiners will be located on skids in the secondary containment). Clarify the type of recombiners to be included in the ABWR design. If portable recombiners are to be used and located outside primary containment, provide information detailing how containment integrity is to be maintained during system operation. Of specific concern is the possibility that leaks in the portions of the system outside containment could result in a flammable mixture and uncontrolled combustion. Additionally, EPRI ALWR requirements presented in section 19B.2.6 require that the “Plant Designer shall define a suitable scheme” for removing residual hydrogen from the containment after an accident. This concern has not been addressed and it appears that there would not be sufficient oxygen in containment after an accident to recombine all the hydrogen. Thus, either an analysis demonstrating that recombiners are sufficient to remove the hydrogen present after a beyond design basis accident, or a method for ensuring that

purging containment, using either the Containment Overpressure Protection System or purging through the Standby Gas Treatment System, will not result in a mixture that would be flammable upon contact with air, is required. (19A.2.12, 19A.2.21, 19A.2.46, 19B.2.6)

430.309

Section 6.2.5 of the ABWR SSAR asserts, but does not demonstrate, that mixing of drywell and suppression chamber atmospheres by natural circulation occurs and would be enhanced by containment sprays. There is no justification for the assertion that combustible mixtures will not form locally. Provide the analysis justifying the assertion that combustible mixtures will not form locally. (19A.2.12, 19A.2.21, 19A.2.46, 19B.2.6)

430.310

There is no consideration of the effects of accidents beyond design basis, as required by TMI issue (TMI-II.B.8). The ABWR SSAR states that there are no design basis events that result in core uncovery or core heatup sufficient to cause significant metal-water reaction, and therefore uses the Regulatory Guide 1.7 design basis metal-water reaction instead of addressing hydrogen generated by a 100% metal-water reaction. Provide an analysis of the capability of the hydrogen control system to mitigate the effects of beyond design basis hydrogen and oxygen generation rates. (19A.2.17, 19A.2.21, 19A.2.46, 19B.2.6)

430.311

The design evaluation of the inerting system uses NEDO-22155 rather than Regulatory Guide 1.7 oxygen generation rates. The NEDO-22155 generation rates are not acceptable for use in licensing submittals. (See the July 6, 1989 NRC SER for NEDO-22155.) Incorporate the appropriate oxygen generation rates in the hydrogen control system analysis.(19A.2.17, 19A.2.21, 19A.2.46, 19B.2.6)

430.312

The containment isolation valves for the Flammability Control System have been identified in Table 6.2-7 of the ABWR SSAR. However, no other information regarding these containment penetrations have been provided. The discussion of the recombiner system to be provided, as part of ABWR SSAR section 6.2.5, should include a description of the dedicated penetrations for this system. Include in your discussion details such as (1) how long after LOCA and at what concentration level of hydrogen, the recombiner has to be activated, (2) line sizes as related to flow requirements, (3) duration of recombiner operation, and (4) interface requirements for referencing applicants with regard to the recombiners (e.g., development of procedural provisions to assure availability of possible shared portable hydrogen recombiners between sites on a timely basis and coordination of surveillance programs in accordance with SRP Section 6.2.5 acceptance criterion II.12). Also, include the flammability control system in SSAR Table 3.2-1. (19A.2.17, 19A.2.21, 19A.2.46, 19B.2.6)

430.313

Regarding additional accident monitoring instrumentation identified in NUREG-0737, TMI Action Item II.F.1, Attachments 1 through 6, discuss compliance with the positions stated in the following clarifications: (19A.2.29)

(1) Clarification (1), (2) and (4) for noble gas effluent monitor (NUREG-Pages II.F.1-2,3 and 6). Note that SSAR Sections 7.5 and 11.5 discuss only some positions identified in the clarifications.

(2) Clarifications (1) through (4) for sampling and analysis of plant effluents as they relate to ABWR scope (NUREG Pages II.F.1-7, 8 and 9); identify any applicable interface requirement for the referencing applicant.

(3) Clarifications (1), (3) and (5) for containment high range radiation monitor (NUREG Pages II.F.1-11 and 13).

(4) Clarification (5) for containment pressure monitor (NUREG Page II.F.14).

(5) Clarification (5) for the containment water level monitors (NUREG II.F.1-16). Include suppression pool water level in SSAR Subsection 7.5.2.1. The staff finds GE’s justification for considering the drywell sump level monitors as Category 3 rather than as Category 1 as required by Regulatory Guide 1.97, Rev 3, unacceptable. Address the above concern.

(6) Clarification (3) for containment hydrogen monitor (NUREG Page II.F.1-18). Also, clarify whether the monitors have the capability to operate from –5 psig to design pressure as required by Regulatory Guide 1.97, Rev 3.

435.63

Description and analysis demonstrating compliance of the offsite power system to regulatory requirements has not been addressed in the ABWR SSAR. Provide a description and analysis demonstrating compliance for the offsite power system within the ABWR standard plant scope from the utility/ABWR interface to the Class 1E distribution system input terminals. Also, provide interface requirements for the offsite power system outside the ABWR standard plant scope from the utility/ABWR interface out to the utility grid system. (19B.2.18, 19B.2.24)

435.64

Provide descriptive information and analysis for reference in the ABWR SSAR, where the descriptive design information or analysis can be found, which demonstrates that the ABWR design is consistent with the ALWR resolution for generic issues described in Section 19B.2.18 and 19B.2.24. (19B.2.18, 19B.2.24)

435.65

It is the staff position that transformers associated with the preferred offsite circuits be separated by the maximum extent practical (preferably on different sides of a building) in order to

minimize the common-cause effects of fire, missiles, or environmental effects on their operation. Provide a description and interface requirements which demonstrates compliance with the staff position. (19B.2.18, 19B.2.24)

435.66

It is the staff position that interconnectors between redundant divisions through safety or non-safety buses shall be maintained with two normally open and interlocked devices that are separate and independent such that single failure or operator error cannot cause the interconnection of or challenge to redundant divisions. Provide a design description, interface requirements, and/or analysis which demonstrates compliance with this staff position. (19B.2.18, 19B.2.24)

435.67

Non-safety computers and transient recorder loads shown on Figure 8.3-5 have provisions included in their power supply design for automatically transferring these loads from class 1E division 1 to 3. In addition, it appears that the power supply may also include provision for automatic transfer of these loads between division 1 and 2. The design does not appear to meet regulatory positions 4b and 4c of Regulatory Guide 1.6 and thus does not appear to meet the independence requirements of criterion 17 of Appendix A to 19 CFR Part 50, the intent of position 1 of Regulatory Guide 1.75, or the intent of generic issue 128. Explain how the design meets the staff requirements or provide design changes such that the ABWR electric system design will meet the independence requirement of criterion 17 or identify this design as being in non-compliance with criterion 17 and provide justification. (19B.2.18, 19B.2.24)

435.68

Identify all safety and non-safety loads that can be powered from more than one Class 1E divisions AC or DC power supplies. (19B.2.18,19B.2.24)

435.69

ABWR resolution (3) of Section 19B.2.24 indicates that alternate AC power for battery chargers is supplied through a series of physically separated breakers from a different division. These interdivisional breakers are in series, mechanically interlocked, and kept normally open during plant operation. Provide justification that this design is more reliable than one where the alternate AC supply comes only from the division the DC system is associated with. In addition, provide physical layout drawings, analysis, and/or interface requirements which demonstrate that when the alternate AC is used the independence and single failure requirements of criterion 17 are still met. (19B.2.18, 19B.2.24)

440.117

Position—The licensees should determine, on a plant-specific basis, by analysis or experiment, the consequences of a loss of cooling water to the reactor recirculation pump seal coolers. The pump seal should be designed to withstand a complete loss of alternating-current (ac) power for at least 2 hours. Adequacy of the seal design should be demonstrated. (19A.2.30)

Resolution—The intent of this item's position is to prevent excessive loss of reactor coolant inventory following an anticipated operational occurrence. Loss of AC power for this is construed to be loss of offsite power.

The staff requires the following additional information from GE to complete the review on this item:

Confirm that failure of the following systems will not generate a LOCA.

(1) Recirculation Motor Cooling System (RMC)

(2) Recirculation Motor Seal Purge System (RMSP)

(3) Recirculation Motor Inflatable Shaft Seal Subsystem (RMISS)

621.1

Identify who performed the ABWR HRA (GE and/or other contractors), and describe the expertise that was included in the HRA team.

621.2

Describe the material and/or analysis that were available and used to support the HRA, including:

(1) Detailed function and task analysis (utilizing the ABWR staffing goals and staffing philosophy);

(2) Procedures or procedure guidelines (draft or preliminary, etc.);

(3) Control room design;

(4) Work station design;

(5) Display design; and

(6) Any other.

Discuss the degree of completeness of each of the materials used in terms of the ABWR design to support the HRA.

621.3

As per Chapter 19 of the SSAR, the HRA methods and procedures identified as used in performance of the ABWR HRA were THERP (Technique for Human Error Rate Predictions, NUREG/CR-1278) and SHARP (Systematic Human Action Reliability Procedure, EPRI NP-3583). Identify which HEPs were derived by each HRA method, and describe any other methods that were used to support these approaches.

621.4

For those HEPs where THERP was used, describe how the Swain and Guttmann Handbook was actually applied in the following areas:

(1) Whether the full analysis methodology was followed;

(2) How base case HEPs were derived;

(3) The data which were used as the source of base case values;

(4) The performance-shaping factors that were applied.

621.5

Chapter 19 (p. 19.3-1) states that the HEPs "were taken predominately from the GESSAR II PRA" and that "most of these values were derived from the Swain and Guttmann Handbook of Human Reliability" which as referenced was published in 1983. However, the GESSAR II PRA was published in 1982, one year prior to the publication of NUREG/CR-1278. In light of this, please identify the version of the Swain and Guttmann Handbook of Human Reliability (NUREG/CR-1278) that was used.

621.6

For those HEPs for which SHARP was used, please provide the documentation called for in the procedure, or, if this approach was not used, please describe how SHARP was actually applied.

621.7

Chapter 19 states that "more recent studies suggest that these values may be somewhat conservative" (p. 19.3-1). Discuss those studies that are used to support this statement, and describe how they apply to ABWR operations.

621.8

As indicated above, Chapter 19 (p. 19.3-1) states that the ABWR HEPs "were predominantly taken from the GESSAR II PRA for which they were collected from various other sources and modified, as appropriate, for the GESSAR application" and that their "application in the ABWR PRA is judged to be acceptable". With respect to this statement, please discuss the following:

(1) The other sources and methods that were used to derive those HEPS. (Reference is made to "the EPRI time-reliability correlation" on p. 19I.4-1— does this refer to the Human Cognitive Reliability (HCR) study?);

(2) If the HCR study was used in support of the HRA, please provide a report of the study to support the evaluation;

621.9

Describe how you accounted, in the HRA, for the use of new, advanced technology in the control room and for the differences in the operator's role in the ABWR vs. a standard control room. That is, how is the operator's role change (due to the introduction of compact work stations and advanced I & C with primary reliance on human-computer interface technology) accounted for in the analysis, with regard to the following:

(1) The appropriateness of the use of numbers from NUREG/CR-1278 for use in the ABWR;

(2) The manner in which HRA subjective judgement was used given the advanced (and different) nature of the control room:

(3) The methods and the experts that were available to modify HEPs for ABWR operations;

(4) Any design features of the ABWR that were used as a basis to lower HEPs which had been obtained from an earlier PRA and, if so used, a discussion of which errors were involved and what technology was assumed to enhance operator performance.

621.10

The introduction of new advanced technology has frequently been associated with the emergence of new human errors. Describe how the ABWR HRA has specifically analyzed the advanced control room, changes in staffing philosophy, etc., to identify potential "new" errors introduced by differences between the ABWR and previous product designs, and which human errors were included in this category. If this has not been done, please discuss your intentions in this regard.

621.11

In summary, given that a variety of source documents were used, please provide an audit trail for each, describing:

(1) The task analysis used;

(2) The PRA which was originally used to provide an HEP;

(3) The method that was used to derive the HEP;

(4) How the HEP was modified for use in subsequent PRAs (such as from Limerick to GESSAR to ABWR), and how design, procedures and operations differences were accounted for;

(5) For which HEPs screening values were used;

(6) Which HEPs were specifically modified for the ABWR.

Please provide the HRA documentation to support the review.

630.1

In item 19A.2.47, NRC position, part (b) you state that you will include information on “the technical resources director by the applicant.” This is either a typographical error or a misinterpretation of the NRC position. It is our position that the information should include “the technical resources directed by the applicant.” Please clarify your statement or provide a basis for the change to the content of the information that we require.(19A.2.47)

725.1

In most of the currently available BWR PRAs, the loss of offsite power sequence with successful recovery of offsite power within 30 minutes (i.e., TM sequence in Fig. 19D.4-4) is transferred to the MSIV closure (i.e., isolation events) event tree. Please provide the basis for transferring it to the reactor shutdown tree (i.e., Fig. 19D.4-1) instead.

725.2

Should not the event tree top event, Q (Feedwater), appearing in the reactor shutdown event tree (Fig. 19D.4-1) be replaced by “Feedwater and PCS”? Otherwise, a branch should be added to the uppermost sequence (with an end state of OK) to determine the success or failure of the top event, W. Note that condenser problems (hardware or others) can lead to a manual shutdown.

725.3

Please provide the basis of not crediting automatic depressurization for the safety function, X, in the reactor shutdown event tree (Fig. 19D.4-1).

725.4

Does ABWR have a design feature which allows the reactor operator to utilize RCIC in steam condensing mode to transfer reactor decay heat to the ultimate heat sink? If yes, why is no credit given to such a feature in evaluating the safety function W (containment heat removal)?

725.5

In essentially all of the event trees shown in Fig. 19D.4-1 through Fig. 19D.4-14, failure of the W function (long term heat removal) is assigned a probability of failing to run RHRA or RHRB or RHRC rather than failure to start and run RHRA or RHRB or RHRC, if the preceding V function (RHR injection or condenser) is a success. This would be correct if one of the RHR pumps was successfully started and run to accomplish the mission of the V function, and then switched to a long term heat removal mode. Notice, however that the success of the V function can also be achieved, as indicated in Table 19.3-2 by using one condenser pump and one condenser transfer pump. In such a case, the approach taken in the ABWR FSR will underestimate the failure probability of W since the RHR pump has to be started and then run throughout the mission time. Also, can one RHR pump alone always accomplish the missions of both the V and the W functions for all the transients including a large LOCA?

725.6

In both the non-isolation event tree (Fig. 19D.4-2) and the isolation/loss of feedwater event tree (Fig. 19D.4-3), the uppermost sequence (with an end state of OK) should branch out at the top event, W, since success of Q (feedwater alone) does not automatically warrant the success of W. The same comment also applies to the IORV event tree (Fig. 19D.4-11).

725.7

In Table 19D.4-1 through Table 19D.4-17, the branch point value of the safety function V (LPFLA or LPFLB or LPFLC available) was assigned a value of 1.27E-03, with the source of the data given as Table 19D.4-1. No such data, however, can be found in Table 19D.4-1. Also, for the loss of offsite power event trees, failure of V (LPFLA or LPFLB or LPFLC or one condensate and one condensate transfer pump) is given a value of 7.37E-03. Again, no such data can be found in the tables. Please explain how this value was calculated.

725.8

For isolation/loss of feedwater events, successful RHR operation using the PCS requires reopening of the MSIVs and the recovery of feedwater if it is initially lost. In Fig. 19D.4-3, which event tree top event takes into consideration the reopening of MSIVs? Also, will the chance of reopening the MSIVs be smaller if there are stuck open SRVs?

725.9

In the loss of offsite power and station blackout event tree (Fig. 19D.4-4), the probability of failing all three diesel generators (7.99E-04) is used to sort out station blackout sequences (i.e., BE2, BE8, and BE0) from the loss of offsite power sequences (i.e., TE2, TE8, and TE0). Note, however, that "all DG not fail" could mean: (1) one DG is available, (2) two DGs are available, or (3) all three DGs are available. In Figs. 19D.4-5 and 19D.4-6, the unavailability of U_h (HPCF B or C with a probability of 1.58E-02) was computed based on the assumption that two diesel generators are available. If only one DG is available at the onset of loss of offsite power, this unavailability could become larger. It appears that some kind of weight-averaging should be applied to modify this value based on the probabilities of having either one or two DGs when the loss of offsite power occurs. Also, in Fig. 19D.4-4, the failure probability of opening SRVs following an ATWS event was taken to be 1.0E-06. For ATWS events, a large number (15) of SRVs need to be opened for pressure relief, and hence, the failure probability of opening the required number of SRVs can be expected to be larger.

725.10

In all of the loss of offsite power event trees (Figs. 19D.4-5, 4-6, and 4-7), the failure probability of HPCF (U_h) is taken to be the same irrespective of the offsite power recovery time and regardless of whether there are stuck-open SRVs. Can the heating up of suppression pool for a prolonged period of time due to stuck-open SRVs adversely affect the availability of HPCP?

725.11

Please provide the basis of not considering stuck-open SRVs in the station blackout event tree (BE2, Fig. 19D.4-8).

725.12

In the same event tree cited above (Question 725.11), the failure probability of W (RHRA or RHRB or RHRC) is taken to be 5.19E-04, which does not correspond to that (1.59E-03) shown in Table 19D.4-1 for the case of loss of offsite power. Are the values shown in the column under the heading of "Loss of Offsite Power" in Table 19D.4-1 also applicable to station blackout? If not, please explain.

725.13

In the station blackout event tree (BE8, Fig. 19D.4-9), why does the sequence with success of RCIC need to be branched out for testing the success of HPCP? According to the success criteria listed in Table 19.3-2, successful core cooling using a high pressure system can be achieved by using either RCIC or one train of HPCF for all transients including loss of offsite power. Furthermore, both HPCF and LPFL require AC power which, in this case, is not available for nearly eight hours. Please explain why both HPCF and LPFL are included as event tree top events.

725.14

For IOR V transients, there is no immediate automatic scram signal, and the operator may be required to manually scram the reactor and start the makeup system before the suppression pool temperature exceeds the heat capacity temperature limit. Please provide the basis of not including "timely manual scram" as an event tree top event in the IOR V event tree (Fig. 19D.4-11).

725.15

Please explain why feedwater (Q) was not credited as a viable means of core cooling in the small LOCA event tree (Fig. 19D.4-12). Note that, according to the success criteria shown in Table 19.3-2, feedwater can be used to successfully cool the core in the event of a small steam LOCA.

725.16

Please explain why HPCF is given credit in the large LOCA event tree (Fig. 19D.4-14) despite the high degree of depressurization caused by the large LOCA.

725.17

Please provide justification of not considering vapor suppression in the large LOCA event tree.

725.18

In constructing the ATWS event tree (Fig. 19D.4-15, no distinction was made between ATWS events with MSIV closure (isolation) and those with bypass available (non-isolation), although the former is generally more severe and limiting. Please explain why the same branch point probabilities were used in quantifying the ATWS sequence frequencies despite differences in the success criteria, such as the time available for the operator to inhibit ADS or the unavailability of normal heat removal system for containment heat removal (see Table 19.3-3).

725.19

It appears that the low core damage frequency (9.lE-09/RY) found for ATWS sequences is mainly driven by the low initiating event frequency (9.34E-09/RY), which was obtained by taking scram failure probability, C, to be 1.0E-08. Please explain in detail how this scram failure probability was calculated. From the fault tree developed for a single control rod drive (Fig. 19D.6-17a, Figure 1), the probability of failure to insert an individual control rod can be estimated to be roughly 3.0E-06. No explanation, however is given as to how this probability is used to generate the probabilities of the basic events shown in the fault tree of control rod drive system (Fig. 19D.6-19a, Figure 1). Also, no probability data is given for the event RPS (RPS fails to initiate scram) appearing in the fault trees for reactivity control (Fig. 19D.6-16b).

725.20

In Table 19.3-3, the time available for the operator to initiate one train of SLC is given to be 10 minutes for both isolation and non-isolation ATWS events. Should not the time available for the former be shorter because the suppression pool is heated up sooner?

725.21

For an ATWS event which is initiated or accompanied by closure of all MSIVs or loss of condenser, can adequate core coolant inventory be maintained by RCIC alone (as indicated in Table 19.3-3)? For some BWRs of current design, such an event requires HPCI or a combination of HPCI and RCIC.

725.22

In quantifying ATWS sequence frequencies, the same branch-point value was used for W (containment heat removal) regardless of whether there are stuck open SRVs. Was suppression pool heating due to stuck-open SRVs taken into account in estimating the failure probability of W?

725.23

Is there any reason why the event tree top event “ADS Inhibit” in the ATWS event tree is placed before “Feedwater or HPCF” and “RCIC” although it appears more logically correct to place it after the latter top events?

725.I4

Was any functional event tree or fault tree developed to analyze the unavailability of feedwater, condensate, and condenser system? How was the unavailability of feedwater (Q), for example, evaluated for different transient initiators?

725.25

In the event tree quantifications, the frequency of a particular accident sequence was obtained by multiplying together the initiating event frequency and the branch point probabilities of the failed safety functions, such as U, V, or W, appearing in the sequence description. This approach is proper if the branch point probabilities were evaluated by properly accounting for the common-mode failures among the event tree top events by linking together the relevant

fault trees. Were these fault tree linkings done in the ABWR analyses to obtain the upper-bound of minimal cut sets for safety function failures such as UV, QUV, or UVW? If not, please explain how the branch-point probabilities were calculated for the individual safety functions such as U, V, or W.

725.26

Were all the system failure probabilities (except for RCIC) listed in Table 19D.4-1 obtained by quantifying the fault trees shown in Section 19D.6? Were the probabilities of failing all ECCS systems computed by linking the high pressure and low pressure system fault trees? If so, which mode of the low pressure system was used? Also, were these values actually used in the event tree quantifications?

725.27

Were the fault trees for the support systems, such as electric power system, service water system and instrumentation system individually quantified? Are the results of such fault tree quantifications (in terms of minimal cut sets) available for comparison with BNL calculations?

725.28

What modifications to the fault tree input data were made to obtain the system failure probabilities corresponding to loss of offsite power (last column of Table 19D.4-1)? Was the failure probability of switchgear taken into consideration when the failure probability of the W function (for example, in Figure 19D.4-7) was calculated?

725.29

Please briefly describe the possible impacts of omitting the development of system fault tree for plant air system on the frontline and the support systems.

725.30

It was noted that a very small fraction of the failure data shown in Table 19D.6-2 through 19D.6-7 are inconsistent with those shown in the relevant fault trees (for example, DIV2MUX, HMV14BHW, and HXV032CQ in Table 19D.6-2). Which values were actually used in the fault tree quantifications?

725.31

The break areas for the various LOCAs (large, medium, and small) are defined to be significantly larger than those used in, for example, the Limerick PRA. Do the initiating event frequencies used in the event tree quantification reflect these changes in the definition of break sizes?

725.32

How does the RWCU (reactor water cleanup) system work to remove decay heat? What suction lines are used? What is the heat sink? Does the non-generative heat exchanger have enough capacity to remove decay heat?

725.33
For RHR shutdown cooling mode, suction is taken from RPV. Where are the points of suction for the three suction lines? Also, where are the discharge points for the core cooling subsystem return lines?

725.34
Questions on Table 19D.4-1.

(1) What modifications were made to the fault trees to obtain the failure probabilities corresponding to large or medium LOCAS?

(2) Are the RCIC failure probabilities calculated by quantifying the revised fault trees in Amendment 8?

(3) What are the failure probabilities corresponding to station blackout?

725.35
What modifications were made to the fault trees to obtain the core damage frequency corresponding to incorporation of (a) gas turbine generator; and (b) fire system water connection?

725.36
Following loss of offsite power, feedwater pumps (motor driven) are tripped and MSIVs are likely to be closed. Are the FW pumps or the RWCU pumps connected to DG power source? Is re-opening of MSIVs considered in calculating the probability of NHR for the W function? In other PRAS, feedwater is considered unavailable following LOOP.

725.37
Class II sequence frequency was calculated to be 4.29E-06. The input to the Class II containment event tree, however, is 2.5E-06. Please explain the difference. Was the CDF for Class II sequences (4.29E-10) obtained by taking 0.01% of 4.29E-06?

725.38
ATWS transient scenarios vary significantly depending on whether MSIV are closed or whether offsite power is available. How can a single ATWS event tree properly handle all ATWS events of different initiators?

725.39
In the ATWS event tree, failure to initiate SLCS is given a probability of 0.2 (time available for the operator = 10 min.) A typical value used for this action in most other BWR PRAs is 0.87 (with time available for the operator = 8 min.). Please explain the difference.

725.40
In the ATWS event tree, the probability of failing to inhibit ADS is taken to be 0.1. A typical value used in other PRAs is 0.5 if high pressure core injection is a failure, and 0.005 if HPCI is

a success. To be able to make such a distinction, the order of the event tree top events for "HPCI" and "failure to inhibit ADS" must be interchanged.

725.41

For loss of offsite power initiators, stuck open relief valves (SORVs) were considered in Amendment 4, but were eliminated in Amendment 8. Please explain why.

725.42

For isolation/loss of FW events, the unavailability of feedwater is taken to be 0.43 (= 40%(l) + 60%(0.05)). Is not the value 0.05 too optimistic for the MSIV closure initiators?

725.43

In order to expedite the staff's review, please provide a copy of the MAAP code and requisite input information that was used in the ABWR evaluation.

725.44

Please provide a copy of the magnetic medium containing all system level fault trees and functional level fault trees modeled for the initiating events applicable to the ABWR.

725.45

Please provide the input files for the MAAP calculations.

725.46

The probability of containment failure resulting from loss of heat removal is given as 3.4E-6 in Section 19.1.2. However, the frequency of containment structural failure resulting from loss of containment heat removal is given as 2.5E-7 per reactor year in Section 19D.5.12.4. Please clarify.

725.47

Is the failure pressure of the upper drywell (UDW) head above 500 F independent of the UDW temperature? If it is a function of temperature, please provide the function. Please also provide the leak area for the high temperature failure. Is high temperature failure considered to be P (penetration) or D (drywell head) failure in the release mode from containment when binning the accident sequences?

725.48

What are the locations and sizes of the passive flooders? Please describe the melting process of the passive flooder fuse including the temperature distribution in the fuse. What is the reliability of these flooders? Are there any examples of their use in other industries?

725.49

The CET for Class IV accidents was not developed because of negligibly low occurrence frequencies (Section 19D.5.11.1). However, CETs for accident classes with similar or lower frequencies (Classes IB-3 AND IIIA) were developed. Please explain.

725.50a

With respect to Firewater Addition (FA), is it necessary to have a separate “FA” category for a mitigating feature? It appears that “FA” is included in “IV” (e.g., Figures 19E.2-6 describe a sequence SBRC-FA-D0. However, this sequence is binned as SBRC-IV-D0 in CET IB-2, Figure 19D.5-8). The CETs do not show any sequences with “FA”.

725.50b

Withdrawn.

725.50c

How is the firewater addition or spray handled in the CETS? It appears that it is included sometimes in “ARV” (e.g., Seq. 3 of CET ID.2) and sometimes in “ARC” (e.g., Seq 6 of CET IA-1 (sic)). Would it not simplify and clarify the CETs if firewater is designated as a separate heading? Firewater spray appears to play a major role in reducing the release fractions by scrubbing in the case of containment failure. (A suppression pool loses its scrubbing function once the vessel fails). Therefore, it is important to know if firewater is available for a particular sequence.

725.51

It is repeatedly stated that corium cools in the LDW after vessel failure by the water which was retained in the lower plenum in many of the accident descriptions. Why did this water not cool corium in the vessel before vessel failure? How much water is available in this manner? Would accidents progress differently if the water cooled the core in vessel?

725.52

Questions on Figures 19E.2-2 (Accident sequence LCLPPFDM)

(1) In Figure C, why does the upper drywell temperature continue to increase throughout the accident?

(2) In Figure E, why does the drywell water level change between the PF opening and the DW head failure?

(3) In Figure B, why does the drywell pressure decrease after water boils away? (The gas temperature does not show any corresponding drop during this period.)

725.53

Referring to Figures 19E.2-5 (Accident Sequence LCHPPFPH): Figure A shows a pressure drop at about 17 hours. This was explained in the text as being due to the flow of water from the suppression pool into the drywell (A similar phenomenon was shown in Figure 19E.2-11.). Please clarify. It appears that the DW pressure should be higher than the WW pressure during this period. This pressure drop appears to delay the DW head failure by about 10 hours. What impact will this have on the final release fraction?

725.54

The suppression pool bypass due to stuck open WW-DW vacuum breakers is of concern only for cases involving wetwell venting. Please explain the consequence Ratio of 825 used in the equation on Page 19E.2-40. In the same equation, the fire water unavailability of 1.5% was assumed, which is considerably lower than 10% used elsewhere. Please explain.

725.55

The CET top event “ARC” (core melt arrest in containment) can occur if any of the following conditions exist, RHR is available, or RHR is recovered, or firewater is available, or PF operates.

Except for firewater, other features are already designated as top events of CETs (CHR, RCH, PF). Is it necessary to have “ARC” as a separate heading? It appears to be duplicative and confusing regarding how “ARC” occurred. (It is confusing since some of the top events are operation/availability of systems while some of them are events caused by operation of the same system.)

725.56

High temperature failure (HTF) occurs if corium is carried to the UDW and no spray is available. Does the probability 0.01 include the probability of both of these occurring? Wouldn’t it be clearer if this heading is replace by “Corium in the UDW” and “Spray Available”? (See also question 725.57a)

725.57

Questions on Class IA/IA.1 and IIIA/IIIA.1 CETs

(1) High temperature failure probability is identical whether RHR is available or not in these CETs. However, if RHR is available, the probability to have UDW spray appears to be higher and, therefore, the probability of high temperature failure smaller. (See the previous question.)

(2) Why isn’t the probability for “ARC Yes” 1.0 when RHR is available (i.e., what does the probability of 1.e-5 represent in Sequence 4 of CET IA?)

(3) Sequence 3 of CET IA is binned as ..FSNN. Does this imply that core melt is arrested in the containment due to FW? Why not RHR?

(4) How is core melt arrested in the containment without RHR for Sequences 4 and 6? Is this due to FW?

(5) What is the basis for the containment failure probability at the time of vessel failure, 0.001, or high temperature failure probability, 0.01? What is the sensitivity of the final consequence to uncertainty in these numbers?

725.58

Questions on Class IB-1/IB-1.1 and IB-3/IB-3.1 CETs

(1) How is the core melt arrested in the containment for Sequences 2 and 4 of these CETs? Are these probability same for IB-1 and IB-3 because they are solely due to FW?

(2) Why isn’t the RHR recovery probability 100% for Sequences 2 and 5 for IB-1?

(3) Why is probability of the RHR recovery failure significantly higher for Sequence 7 than for Sequence 4 in IB-1?

(4) Why is the probability of RHR recovery failure 5 times higher for Sequence 4 of IB-3 than Sequence 4 of IB-1, while they remain the same between Sequences 7 of IB-1 and IB3? (Incidentally, the “RCH No” branch probability for Sequence 7 of IB-3 appears to be misprinted. It should be 0.1, not 0.01.)

(5) Sequence 7 of IB-1 is binned as PFDH while Sequence 7 of IB1.1 as PSDN. This implies that the consequence of the low pressure vessel failure is more significant than that of high pressure. Please explain. (The same question for IB-3.)

725.59

Questions on Class IB-2 CET

(1) The core damage frequency for this class is not the same as that of Table 19.3-6. Please clarify which is correct.

(2) The probability of failure to depressurize the reactor is 3 times lower for Class IB-2 compared to Class IB-1/3 (0.002 vs. 0.006). Is this due to the time available before depressurization? Does this probability depend on how much time is available before the demand of this equipment? (i.e., what action can be taken to improve availability of this equipment before challenge regardless of how much time is available?)

(3) Please provide the basis for the “ARV No” branch probability of 0.006 for Sequences 4 to 7 and 0.6 for Sequence 12.

(4) Why is the “ARC No” branch probability of Sequence 7 significantly higher for this CET than others (0.05 vs. 0.01)? Why isn’t this branch further divided depending on the RHR recovery? (This is done for cases which have even smaller probabilities.)

(5) Sequence 6 is binned as FSDH. This is the only place where a sequence is binned as “High” when FW scrubbing is available. Please explain.

(6) Why is RHR unavailability significantly lower for Sequence 11 compared to the similar sequences for other CETs (0.01 vs. 0.05 for IA)?

(7) Why isn’t Sequence 12 further branched like the similar sequences of IB-3.1?

725.60

Questions on Classes ID and IIID CETS.

(1) How is core melt arrested in RPV? In this solely due to FW? (This branch existed in Amendment 4 which did not have FW.)

(2) Why is the probability of RHR recovery failure significantly higher in this CET than in others?

725.61

Questions on CET II

The “CC No” branch fraction is significantly reduced from Amendment 4 to Amendment 8 (0.001 from 0.1). Besides the availability of firewater, what else contributed to this reduction?

725.62

According to Response 5 of GE’s response to previous staff questions, all the Residual Heat Removal (RHR) pumps will start automatically upon receipt of low water level signal or high drywell pressure signal and can be transferred to other operating modes while they are running. Is the transfer of the RHR pump flow from injection mode (referred to “V”) to the containment heat removal mode (referred to “W”) done automatically without requiring any operator actions? If so, provide discussions regarding modeling aspect of operator actions for the containment heat removal mode of the RHR system.

725.63

For scenario involving vessel isolation event followed by the failure of the High Pressure Core Flooders (HPCF), Reactor Core Isolation Cooling (RCIC) System and RWCU System, and successful vessel depressurization, will both “V” function and “W” functions be required simultaneously for successful core cooling (during the mission time considered) and long-term heat removal? If so, state the minimum trains of the RHR system needed to avoid a core damage.

725.64

By definition of “Class 2 Sequences,” the containment heat removal systems (RHR system) have failed following a transient and a postulated LOCA event. Therefore, provide discussions regarding adequacy of crediting the RHR system (such as fast recovery) for the scenario involving a vessel isolation event followed by the failure of the HPCF system, the RCIC system, and successful vessel depressurization with coolant injection only achievable by the LPFL mode of RHR. If the RHR system can be used (during this scenario) for both “V” and “W” functions, can train A of the RHR system alone perform both “V” and “W” functions to avoid a core damage?

725.65

The staff notes that the pumping capacity of the RHR pumps of the ABWR design is lower than that of the operating BWR designs. Therefore, provide discussions regarding the modeling adequacy of the RHR system (use of one of three RHR trains to maintain the pool temperature below the heat capacity-temperature limit) for the scenario involving the vessel isolation event followed by a fail-to-scram event. GE's discussions should include supporting pool temperature calculations, including the assumed amount of heat dump to the pool following the above scenario.

725.66

The staff believes that a gas turbine-generator (in addition to the three train diesel generator system) added to the ABWR design will reduce the frequency of sequences involving early core damage following a loss of offsite power event with a postulated common mode failure of the diesels. Thus, provide discussions for the following:

(1) What is GE's definition for the black-start capability for the gas turbine-generator?

(2) Will the gas turbine-generator be started automatically?

(3) If a start failure of the gas turbine-generator will occur, can it be started from the main control room?

(4) Does the operator have to decide as to which class IE 4.16kV bus should receive AC power generated by the gas turbine-generator?

(5) Did GE perform a trade-off study involving the benefits of a seismically qualified gas turbine-generator?

(6) What are the assumptions made in quantifying the results provided in Table 19.3-6 of the ABWR PRA (Amendment 9) which includes the impact of adding a gas turbine-generator? In particular, were the initiating event frequencies (such as Be2, Be8, Be0, Te2, Te8, Te0) recalculated by modifying the event tree provided in Figure 19D.4-4? If so, provide these estimates. Also, provide, for the case of adding a gas turbine-generator, similar results provided in Tables 19D.4-1 and 19D.4-3.

725.67

Provide discussions related to the use of the RCIC system unavailability estimate documented in Table 19D.4-1 (under the column of offsite power event), in the event tree quantification. Also, provide statements related to the consistency of the RCIC system unavailability estimate used for the quantification of the ATWS event tree and the corresponding estimate documented in the Table 19D.4-1.

725.68

Provide the scientific details of seismic hazard analyses performed for the ABWR design review and the basis for selection of the seismic hazard curve (Figure 19.4-2). The discussion

should include site seismicity characterization of various (five reference sites) sites considered in eastern United States of America, including the combination method used to develop a single enveloping seismicity hazard curve to represent an enveloping site to locate the ABWR design, and the associated uncertainty estimates for the use of a single seismicity hazard curve. The discussion of the site characterization should include critical site parameters such as soil-structure interaction for various sites considered. There are some seismic terms used in GE's seismic risk analysis which are confusing to the staff. What is the parameter used for describing the seismic hazard and fragility? For example, it is variously used to represent as the effective peak ground (Figure 19.4-2) and mean peak ground acceleration (Section 19.4.3.2.1).

725.69

Provide the ABWR-specific fragility calculations for the following structures and components: Containment, Reactor Building, Main Control Room (including control room suspended ceilings, if any), Reactor Pressure Vessel (RPV) RPV Pedestal, RPV Shroud Support, CRD Guide Tubes, CRD Housings, Fuel Assemblies, Containment Vent System, Passive Flooder, SRV Pipes to Suppression Pool. If generic component fragilities have been used, provide a detailed discussion how the generic component fragilities were assigned. The discussion should include also applicability of the uncertainty estimates due to variations in ABWR design-specific component design may have.

Does the failure mode, "Relay Chattering," applicable to the ABWR design? If so, provide discussions regarding the modeling of electrical equipment (such as breaker) to account for relay chattering effect in fragility quantification. Provide also discussions regarding sequences (such as loss of containment isolation function) that could result from relay chattering failure mode, and method of quantifying such failure modes (including human recovery actions involved, if any).

Provide the details regarding the seismic capacity of the fire protection system (including the valves F005, A, B, C of the ac independent fire water system. Provide also the seismic capacity of small piping (if used) and valves (14 and 22 inches in size) of the containment overpressure relief (COR) system, addressing the failure mode, "Normally open valves fail closed" and including human recovery actions involved, if any.

725.70

The staff understands that the seismic PRA performed for the ABWR design is limited in nature due to the design stage (FDA). However, our past seismic risk review experience indicates that seismic risk profiles of as-built-ABWR plant in U.S. could be different due to various in construction standards by various architects. Therefore, provide discussions regarding the construction interface requirements such as allocated fragility estimates for all applicable mechanical and electrical component of the ABWR design, as practicable, including the severe-accident design basis and/or goals on which allocation of such fragility estimates will be performed. These discussions should also include consistency between requirements outlined

in Electric Power Research Institute (EPRI)—Advanced Light Water Reactor (ALWR) Requirements Document, and design requirements to be proposed to various architects by GE.

725.71

Provide ABWR-specific layout drawings (in larger size) which show clearly major structures and equipment. Provide also as-designed structural drawings which show the details of the RPV support arrangement, RPV internals arrangement, drywell and the reactor building.

725.72

Provide a copy of the ABWR PRA seismic input data such as seismic hazard curve and seismic accident sequences applicable to the ABWR design in the form of a hard copy (tabular forms and boolean equations) as well as a magnetic media. These data are needed to facilitate staff's audit review.

725.73

The staff believes that the determination of a particular seismic intensity (for risk modeling purposes) at which evacuation scheme at a particular site following a postulated severe-accident will impact greatly the risk estimates (early fatality estimates). Provide discussions regarding the determination of the break point of the seismic intensity (in terms of EPG) at which evacuation were considered impossible for ABWR risk estimation purposes.

725.74

Our past PRA review experience indicates that fires and internal floods contribute significantly to the overall core damage frequency at nuclear power plants. The staff also believes that, with respect to the ABWR design protection against fires and internal floods, GE will provide significant design improvements to current separation requirements and divisional (redundancy) requirements related to all safety systems and components. Nevertheless, the ABWR PRA (Amendment 9) has not documented the core damage frequency analysis of fires and internal floods. Therefore, provide the results of screening analysis (including the screening criteria) performed for the ABWR design to show that fires (panel fires, transient combustible fire, cable fires) and room-specific floods do not significantly contribute to the overall core damage frequency. Provide also statements regarding consistency between requirements outlined in the ALWR Requirements Document and current ABWR design requirements related to fire protection and flood protection schemes.

725.75

In developing the fault trees for seismically induced failure of the ECCS, such as HPCF, RCIC, LPCF and RHR (Figures 19I.2-1 through Figure 19I.2-4 of the ABWR PRA), no explicit modeling of the dependence of these ECCS on electric power or service water system was made. Nevertheless, fault trees were developed in Figure 19I.2-6 and Figure 19I.2-7 to depict seismically induced failure of Division 1 service water and seismically induced failure of Division 1 electrical power respectively. Please explain how the latter two fault trees developed for the support system were combined with event tree top events to generate minimal cut sets for seismic core damage sequences.

725.76
Following loss of offsite power due to seismic events, an important subsequent concern is whether or not emergency power and service water are available. Failure of emergency power (diesels or gas turbine generator) and failure of service water system may be considered as two virtually independent events. In the seismic event tree (Figure 19I.3-1), however, these two events are combined together and treated as a single event tree top event, PW. Please explain how the failure probability of this top event was estimated. Was the gas turbine generator included in evaluating the availability of emergency power?

725.77
Were random failures of the ECCS, such as HPCF, RCIC, LPCF and RHR, taken into account in the quantifications of seismic core damage frequency? If so, please provide a list of random failure probabilities for the important systems and components used in the quantifications.

725.78
On page 19.4-11 of the ABWR PRA (second paragraph), it is stated that "Since these fault trees (meaning those shown in Appendix I) are specifically for evaluation of seismically-induced failures, only those components vulnerable to seismic failure are included in the trees." In reality, however, those fault trees also contain basic events (depicted with an "X"), that would not occur as a result of an earthquake. Please explain the contradiction.

725.79
Please provide justification of considering heat exchanger failure in the RHR and service water fault trees, while ignoring it in the fault trees of RCIC, HPCF and LPCF. 13. In the fault tree developed for service water system (Figure 19I.2-6), the motor-operated valve, WMVS3DH, is considered seismically vulnerable, while three other similar motor-operated valves are considered seismically invulnerable. What is the basis for making such a distinction? In the fault tree depicting seismically induced failures of RCIC (Figure 19I.2-2), three identical basic events are used to denote non-seismic failure of an isolation valve (MOV). Are these three basic events intended for failures of three different isolation valves?

725.80
Please provide a terse but systematic description of how the Boolean expressions derived from the seismic event trees and fault trees are combined with seismic hazard function, component and structure fragilities and other unavailability data, and integrated to obtain the frequency of individual accident sequences.

725.81
For ATWS events with failure to initiate SLC, what alternative means are available for injecting boron in order to shut down the reactor? What failure probability was used in the sequence frequency quantification for the event tree top event, FCTR (flow control/alternate boron), appearing in Figures 19I3-1, 3-3 and 3-4?

725.82

The event tree top event, W1, appearing in Figures 19I.3-2, 3-3 and 3-4, is defined to be "at least one RHR." How many trains (1, 2 or 3) of RHR were actually used in the sequence frequency quantifications? Please also list the random failure probability assigned to this event in each figure.

725.83

In the seismic event tree, Figure 19I.3-1, credit is given to fire water (event tree top event, FA) for the following transient scenarios: (a) station blackout, successful scram, failure of RCIC; (b) station blackout, successful scram and RCIC; and (c) station blackout, failure of scram but successful RCIC. What is the unavailability of fire water system in each case?

725.84

In the seismic ATWS event tree, Figure 19I.3-3, the last sequence involves failure of SRVs to open following the inception of an Loop ATWS. Please explain why this sequence is classified as Class IC, which, by definition, involves low pressure vessel failure. Please also clarify the description of accident classes for Class IV-1 (ATWS with one injection pump) and Classes IV-2, 3, 5 (ATWS with multiple injection pumps) in connection with the relevant sequence classifications performed in Figure 19I.3-3. What is the basis of choosing 2, 3 or 5?

725.85

The suppression pool drain accidents due to RHR pipe break are considered to be 00SN, which implies no fission product release. However, if the suppression pool is drained, the passive flooder is not operable, and therefore extensive CCI will continue. Why is this effect not considered in determining the fission product release for this sequence?

725.86

The firewater availability is considered to be 0.9 for the vessel cooling except for the Class IB-2 accidents, where it is 0.999. The firewater availability for the drywell spray is also assumed to be 0.999 (page 19J.4-1). However, the firewater availability in the internal event analysis was assumed to be 0.9 for vessel cooling and 0.99 for the drywell spray. Why are these substantially more reliable for the seismic events?

725.87

On page 19J.3-1, it is stated that "ARC" is solely due to firewater. Why then is the "ARC" Yes branch fraction not 0.999 in the CET's (why 0.944)?

725.88

How are the "CHR" branch fractions evaluated?

725.89

In Figure 19J.5-7, Sequence 4 was binned as NSRCFSDL. Shouldn't this be binned as OK, since this sequence represents continued core cooling by firewater? (Compare this with Figure

19J.5-6 for Class II.) Why are the “CHR” No and “CC” Yes branch fractions of Class IV not same with those of Classes II?

725.90

The firewater availability is assumed to be 0.9 for Class IV (Figure 19J.5-7). However, it was stated in the internal event analysis that no credit was taken for the firewater system to prevent core damage for Class IV because the stability of the reactor during an ATWS has not been examined (page 19D.5-10). Please clarify.

725.91

It appears that the loss of transformer contributes significantly to “loss of core cooling” accidents. Why isn’t this considered to be a blackout sequence (IB)? What fraction of Class IA is due to this scenario and what fraction is due to other causes such as loss of injection pumps or lines, etc.? What is the RHR recovery probability for each of these sequences?

725.92

It is stated on page 19J.4-1 that the reliability used for the firewater system is also used for the transformer bypass operation. Does the “ARC” Yes branch fraction take this high reliability into consideration?

725.93

Why is RHR assumed to be lost for the Class IA accidents? (In the internal events CET for Class IA the RHR availability was 0.99.) Do all accidents in SCET assume loss of power due to loss of transformer and require the bypass of the transformer?

725.94

Is the loss of the offsite power by seismic events with subsequent failure of onsite power considered to be IB-2? Does Class IB-2 include the loss of power due to loss of transformers? What is the RHR recovery probability for each of these sequences?

910.26

(1) In Appendix 19B.2.4 of the Standard Safety Analysis Report, ABWR resolution of GSI A-29 relies upon separation of trains and usual access constraints and security provisions for protection against insider sabotage. Separation of trains has benefit against an external adversary, but it does not assure additional protection against an insider with authorized access to both trains. Card-readers on doors between trains could present a deterrent effect against insider sabotage, but locked interior doors also have the potential for interfering with rapid access under emergency conditions. Separation of trains also allows the possibility of plant management to group plant personnel into teams that each work on only one train (such as used to enhance quality of operations at the BWR-6 in Cofrentes, Spain. No information is presented on how security access controls would be used to deter insider sabotage at an ABWR, nor to assure that these controls would not interfere with safety. Furthermore,

Appendix 19B.2.4 of the ABWR Standard Safety Analysis Report does not identify design features that decrease reliance on physical security programs for protection against an insider.

(2) Appendix 19B.2.4 does not contain a commitment regarding the ABWR resolution of GSI A-29. Of the 13 pages devoted to GSI A-29, about 12 pages present BWR requirements of the draft EPRI-ALWR Requirements Document, most of which are not relevant to GSI A-29. Although the staff has some as yet unresolved questions about the EPRI requirements, the section titled “ABWR Resolution” does not commit to meeting these draft requirements. It states the ABWR is “considered to comply with” and “is consistent with” the ALWR requirements. It also states that the ABWR complies “with 10 CFR 73 and therefore with the ALWR requirements,” whereas compliance with existing 10 CFR Part 73 neither implies compliance with the somewhat different ALWR requirements nor commits to any improvements over existing designs that also comply with 10 CFR Part 73. Chapter 9, Section 5.2.2.1 of the ALWR Requirements Document is quoted on page 19B.2-11 as:

“. . . a sabotage vulnerability analysis shall be conducted prior to finalizing design. Such analysis shall be in accordance with the criteria and assumptions in Section 5.1.3.”

Further, Section 5.2.4.2 (on page 19B.2-12) states:

“The design of the security system shall include an evaluation of its impact on plant operation, testing, and maintenance.”

Neither of these ALWR required evaluations is contained in the ABWR Standard Safety Analysis Report.

In addition. ABWR Response 910.10 says that issues dealing with plant internal security communications are outside the scope of the ABWR Standard Plant design, although the ALWR Requirements Document cited in Appendix 19B.2.4(19) includes such a communication requirements.

(3) Response to a staff request for additional information are given on pages 20.3-200 through 20.3-202 of the ABWR Standard Safety Analysis Report. Several of these responses are inadequate or incorrect. For example, Response 910.7 incorrectly implies that Appendix 19B.2.4 includes a discussion of the insider and outsider sabotage actions that would be necessary to cause significant core damage or Part 100 release levels.

(4) The Commission’s Severe Accident Policy Statement included the statement:

“The Commission also recognizes the importance of such potential contributors to severe accident risk as human performance and sabotage. The issue of both insider

and outsider sabotage threats will be carefully analyzed and, to the extent practicable, will be emphasized in the design and in the operating procedures developed for new plants.'

To comply with this position and to resolve GSI A-29, the staff anticipated that a systematic analysis of sabotage vulnerabilities would be performed to identify combinations of components which, if tampered with, would lead to core damage. This would have allowed an examination of the results to see if design changes were feasible to ensure that the tampering would be detected before the entire set could be disabled, or to see if it was feasible to "harden" critical components to make them less susceptible to tampering. Such an analysis has not been presented. (19B.2.24)