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

October 1, 1992

Docket No. 52-003

Mr. Nicnolas J. Liparulo
Nuclear Safety and Regulatory Activities
Westinghouse Electric Corporation
P.O. Box 355
Pittsburgh, Pennsylvania 15230

Dear Mr. Liparulo:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION ON THE AP600

As a result of its review of the June 1992 application for design certification of the AP600, the staff has determined that it needs additional information in order to complete its review. The additional information is needed in the areas of mechanical engineering (Q210.1-Q210.26),* structural engineering (Q220.1-Q220.26), seismic design (Q230.1-Q230.23), site characteristics (Q231.1-Q231.14), inservice inspection (Q250.1-250.2J), component integrity (Q251.1-Q251.32), materials engineering (Q252.1-Q252.145), chemical engineering (Q281.1-Q281.19), and radiation protection (Q471.1-Q471.3). Enclosed are the staff's questions. Please respond to this request within 120 days of the date of receipt of this letter.

You have requested that portions of the information submitted in the June 1992 application for design certification be exempt from mandatory public disclosure. While the staff has not completed its review of your request in accordance with the requirements of 10 CFR 2.790, that portion of the submitted information is being withheld from public disclosure pending the staff's final determination. The staff concludes that this request for additional information does not contain those portions of the information for which exemption is sought. However, the staff will withhold this letter from public disclosure for 30 calendar days from the date of this letter to allow Westinghouse the opportunity to verify the staff's conclusions. If, after that time, you do not request that all or portions of the information in the enclosures be withheld from public disclosure in accordance with 10 CFR 2.790, this letter will be placed in the NRC's Public Document Room.

*The numbers in parentheses designate the tracking numbers assigned to the questions.

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The reporting and/or recording requirements contained in this letter affect fewer than ten respondents; therefore, OMB clearance is not required un'er P.L. 96-511.

If you have any questions regarding this matter, you can contact n at (301) 504-1120.

Sincerely,

Original Signed By:

Thomas J. Kenyon, Project M: ager
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Associate Director for Advanced Reactors
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Enclosure:
As stated

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Mr. Nicholas J. Liparulo
Docket No. 52-003

Westinghouse Electric Corporation
AP600

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ENCLOSURE
REQUEST FOR ADDITIONAL INFORMATION
ON THE WESTINGHOUSE AP600 DESIGN

MECHANICAL ENGINEERING

Chapter 3

210.1 Discuss the justification for the safety and seismic classification of structures, systems, and components (SSCs) that are unique to the passive design of the AP600 (i.e. Passive Core Cooling System, Passive Containment Cooling System, etc.). The staff is concerned that there is no previous experience with these systems and they fall outside the structures, systems, and components that have traditionally been classified utilizing the guidelines set forth in Regulatory Guides 1.26, and 1.29 for safety and seismic classification, respectively (Section 3.2).

210.2 The discussion in Paragraph (2)(x) of Section 1.9.3 of the SSAR relative to testing of relief and safety valves in accordance with Item II.D.1 of NUREG-0737 requires additional details. The staff concludes that Westinghouse should add the following commitment to this discussion or provide justification for not doing so:

> All of the reactor coolant system relief and safety valves and their associated discharge piping in the AP600 design are similar to those items that were tested by EPRI and documented in Reference 2. Any plant specific relief and safety valves and discharge piping that are not similar to those tested by EPRI will be tested by the holder of a Combined Operating License in accordance with the guidelines of Item II.D.1 of NUREG-0737.

210.3 Section 3.2.1.1.2 of the SSAR references Section 3.7 for the criteria used for the design of seismic Category II structures, systems, and components. In Section 3.7.3.13.3, "Interaction of Other Piping with Seismic Category I Piping," one of the analysis options for seismic Category II piping systems is "enveloping methods that limit stresses to the level D limits of Equation (9), ND-3653 of the ASME Code, Section III."

a. Provide a brief description of "enveloping methods."

b. It is the staff's understanding that, to be consistent with the definition of seismic Category II in Section 3.2.1.1.2, the loads resulting from an SSE will be used in the Equation (9) calculation which will then be compared to the ASME Service Level D limits in ND-3655 of ASME Section III. If this interpretation is not correct, provide a description of how this criteria will be implemented.

210.4 The last sentence of the first paragraph in Section 3.2.2.1 of the SSAR states that "These definitions are consistent with the draft ANS Definitions for LWR Standards." These definitions should be explicitly identified in the SSAR since the staff does not presently endorse an ANS standard for the classifications of structures, systems, and components. The staff relies on Regulatory Guide 1.26 for that purpose. Provide technical justification for any deviations from Regulatory Guide 1.26.

210.5 Table 3.2-3 of the SSAR contains safety classifications for eleven heating, ventilation, and air conditioning (HVAC) systems, some of which contain safety-related components. HVAC ductwork and its supports are not included in this table. However, Section 9.4.1 of the SSAR states that portions of one of the subsystems of the nuclear island nonradioactive ventilation system are safety-related, and Sections 9.4.6 and 9.4.7 state that some equipment, ductwork and supports in the containment recirculation cooling and the containment air filtration systems are designed as seismic Category II.

a. If the AP600 design contains any safety-related HVAC ductwork and supports, add these items to Table 3.2-3 of the SSAR. Provide the principal construction code for both ductwork and supports.

b. In Sections 9.4.6 and 9.4.7 of the SSAR, the "SMACNA HVAC Duct Construction Standards - Metal and Flexible," 1985, are referenced for design, testing and construction requirements of seismic Category II ductwork and supports. These standards do not contain any seismic design criteria and are not applicable to supports. Therefore, they are not completely acceptable for the design of seismic Category II items. Provide additional criteria for these items. In addition, add these items to Table 3.2-3 of the SSAR.

210.6 Section 3.6.2.1.1 of the SSAR states that "Breaks are not postulated in these sections of pipe, including the reactor coolant loop and pressurizer surge line, that meet the requirements for mechanistic break as described in Subsection 3.6.3." If the pipe cannot meet the limitations and acceptance criteria for the leak-before-break methodology as discussed in Paragraph II.D of Enclosure 1 to the draft Commission paper, "Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements," February 27, 1992, excluding high- and moderate-energy piping from the guidance of Section 3.6.2 and Branch Technical Position MEB 3-1 of the SRP is not acceptable. Identify the high- and moderate-energy pipe that will not meet this acceptance criteria and evaluate it in accordance with the guidance of Section 3.6.2 and Branch Technical Position MEB 3-1 of the SRP (see Q252.2-Q252.14).

210.7 Section 3.7.3.2 of the SSAR, "Determination of Number of Earthquake Cycles," states that for cyclic motion due to earthquakes smaller than the SSE, subsystems sensitive to fatigue are evaluated by assuming two seismic events, each resulting in 10 full-stress cycles with magnitude equal to 50% of the calculated SSE response for structures and components. Discuss the technical justification for the selection of these values for the AP600 design.

210.8 Section 3.7.3.8.2.2 of the SSAR states that for ASME Class 1 piping equal to or less than one inch nominal pipe size and ASME Class 2 and 3 piping equal to or less than two inch nominal pipe size, one of the following three methods of analysis may be used:

a. The method for large diameter pipe described in Section 3.7.3.8.2.1 of the SSAR.

b. Equivalent static analysis.

c. Seismic qualification by experience based on the guidelines in EPRI Report NP-6628, "Procedure for Seismic Evaluation and Design of Small Bore Piping."

If the procedure for use of the equivalent static analysis as noted in Item b above is different from that described in Section 3.7.3.5 of the SSAR, revise Section 3.7.3.8.2.2 to provide a detailed description of the methodology to be used.

The staff is currently reviewing EPRI NP-6628 as a topical report, which was submitted to the staff by the Nuclear Management and Resources Council in a letter dated March 19, 1991. Pending completion of this review, the staff's position is that the methodology in this report is not acceptable. Revise Section 3.7.3.8.2.2 to remove the reference to EPRI NP-6628.

210.9 Section 3.7.3.8.3 of the SSAR, "Piping Systems on Modules," states that modules are constructed using a structural steel framework to support the equipment, pipe, and pipe supports in the module and that, with one exception, the framework is designed as part of the building structure. If, subsequent to installation of the modules, the framework is relied upon to support any portion of the piping, provide the basis for not complying with the jurisdictional boundary rules in Section NF of Section III of the ASME Code.

210.10 Provide the basis for all of the criteria to be used to decouple the analyses of the structural frame from that of both the supported and supporting piping in Section 3.7.3.8.3 of the SSAR. In addition, for those analyses that are not decoupled, provide a description of how the interaction between the structural framework and the piping will be incorporated into the analysis.

210.11 Clarify the discussion in Section 3.7.3.9 of the SSAR on the use of the independent support motion (ISM) method of modal analysis of piping systems to address the following concerns:

a. The proposed ISM method is inconsistent with the recommendations in Sections 2.3 and 2.4 of NUREG-1061, "Report of the USNRC Piping and Review Committee," Volume 4. Provide further technical justification. As a part of these recommendations, a support group is defined by supports that have the same time history input. This usually means all supports located on the same floor (or portions of a floor) of a structure.

b. The damping values in Section 3.7.1 of the SSAR are referenced for use with the ISM method. This implies that the AP600 design incorporates the use of ASME Code Case (CC) N-411, "Alternate Damping Values for Response Spectra Analysis of Classes 1, 2, and Piping, Section III, Division 1" in conjunction with the ISM method. One of the conditions in RG 1.84, "Design and Fabrication Code Case Acceptability, ASME Section III, Division 1," relative to the use of CC N-411 is that the staff's acceptance of the use of the damping values in CC N-411 with the ISM method is pending further justification. Since the proposed ISM method is not in accordance with the recommendations in Item a above, provide further technical justification for this approach.

210.12 Section 3.7.3.9 of the SSAR states that the results of the modal spectrum analysis (multiple input or envelope) are combined with the results from seismic anchor motion (SAM) by the square root sum of the squares method (SRSS). The ASCE Standard 4-86, "Seismic Analysis of Safety-Related Nuclear Structures and Commentary on Seismic Analysis" is referenced as the basis for this criterion. The staff has not endorsed ASCE Standard 4-86, and does not completely agree with this criterion. For the multiple input or ISM method of analysis, this criterion may lead to unconservative results in some cases. The staff's position, as given in Section 3.9.2.II.2.g of the SRP, is that the responses due to the inertia effect and seismic anchor motion should be combined by the absolute sum method. Provide the technical justification for combining the modal spectrum analysis results and the SAM results using the SRSS method.

210.13 Section 3.7.3.15 of the SSAR, "Analysis Procedure for Piping," references the information in Section 3.7.1.3, Table 3.7.1-1, and Figure 3.7.1-13 for certain damping values. For the primary coolant loop and other piping systems, ASME Code Case N-411 is referenced in Table 3.7.1-1. Add a note to Table 3.7.1-1 which states that the damping values in Code Case N-411 can be used only as conditioned by RG 1.84. In addition, provide the basis for the 20% damping value which is listed in Table 3.7.1-1 and Figure 3.7.1-13 of the SSAR for 50% to fully loaded cable trays and related supports.

210.14 Section 3.7.3.15 of the SSAR states that piping systems, including coupled equipment, valves and structural frames, can be evaluated with Code Case N-411 damping. Provide the basis for using Code Case N-411 damping values for structural frames.

210.15 The guidelines of Paragraph II.2 in Section 3.9.1 of the SRP state that a list of computer programs that will be used in dynamic and static analyses to determine the structural and functional integrity of seismic Category I Code and non-Code items, and the analyses to determine stresses should be provided. Provide such a list. Also, discuss the various programs' applicability and validity. At present, Section 3.9.1.2 of the SSAR only references the quality assurance program (as described in Chapter 17 of the SSAR) for this information.

210.16 As indicated in Section 3.9.2.3 of the SSAR, the reactor vessel internals in the AP600 are similar in size and configuration to the 3-loop reactor at the H.B. Robinson plant with additional design changes from several reference reactors. However, the AP600 is not a 3-loop reactor, and effects of those design changes, although their acceptability were individually verified by separate tests in different reactors or lab conditions, may interact and result in unacceptable dynamic response. Since flow-induced excitations are complex and sensitive to a simultaneous effect of several parameters, such as configuration of flow path, pressure, temperature, flow velocity, etc., provide details of the evaluation to show how a combination of analysis, testing, and comparison to the results in several reference plants was used to verify the acceptability of flow-induced vibrations of the internals under operational transients and steady-state conditions. In addition, describe acceptance criteria and verify that the above stated evaluation, including detail drawings and calculations, was properly documented.

210.17 Provide configurations and key dimensions of major components of the reactor internals of the standard design to verify that the analytical models were accurately constructed (Section 3.9.2.3).

210.18 Since the AP600 design has different coolant loop configuration from the design of H.B. Robinson plant (see Q210.16), and it also has incorporated additional design changes from several reference plants, it is difficult to visualize the assertions that the reactor internals of the H.B. Robinson design is the valid representative for the AP600 internals. A vibration measurement program should be implemented per RG 1.20 during the preoperational test for either the first AP600 internals or the internals similar to the AP600 but with some design modifications (the Non-prototype Category II). Provide detailed information regarding the vibration measurement program, including numbers, types and locations of sensors, the basis of sensor selection and analyses for predicting levels of response of individual sensors. In addition, acceptance criteria of vibration measurements should also be described (Section 3.9.2.4).

210.19 As indicated in Section 3.9.2.5 of the SSAR, design limitations established for the internals consist of stress criteria and considerations over deflection and stability. Provide quantified details of such limits.

210.20 Section 3.9.2.5 of the SSAR indicates that leak-before-break (LBB) methodology is being applied to the reactor coolant system piping of four-inch nominal pipe size or larger. Thus, the conditions evaluated for dynamic effects of pipe rupture are only based on mechanistically postulated breaks in smaller lines. According to rules published in

Federal Register Notice, Vol. 52, No. 167, dated August 28, 1987 and as discussed in Paragraph II.D of Enclosure 1 to the draft Commission paper, "Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements," February 27, 1992, application of LBB is plant-specific and system-specific and requires NRC approval on case-by-case basis. Explain how the faulted condition analysis of AP600 reactor internals complies with the above stated regulation and guidance.

210.21 Verify availability of analysis performed for demonstrating design adequacy of the AP600 reactor internals to withstand the loads from a pipe break in combination with the SSE. This analysis is essential to ensure structural integrity and operability of the internals for the faulted conditions. Provide details of such an analysis (Section 3.9.2.5).

210.22 The design of the guide tubes was based on pipe break sizes consistent with the application of the LBB methodology. Since the outcome of the staff's review of the application of LBB to the AP600 design is uncertain, the staff recommends that Westinghouse discuss how the design will ensure the function of the control rods if the criteria of Section 3.6.2 and BTP MEB 3-1 is used to determine pipe break size (Section 3.9.2.5).

210.23 Section 3.9.2.6 of the SSAR indicates that the results of dynamic analysis of reactor internals have been compared to the results of preoperational testing in reference plants. Describe the analytical model used and provide details of the comparison.

210.24 Section 3.9.6 of the SSAR states that an inservice testing (IST) program for pumps and valves will be submitted by the Combined License (COL) applicant. However, there is no mention of a submittal of an IST program by Westinghouse for the AP600 design certification application.

Provide an IST program to demonstrate that adequate design and access provisions will be incorporated to permit the effective performance of IST. The staff will review this IST program to ensure that the Westinghouse's commitments regarding the ability to test pumps and valves can be met.

210.25 Section 3.9.6 of the SSAR, particularly Section 3.9.6.2, indicates that only ASME Code Class 1, 2, and 3 passive safety-related valves will be included in the IST program for the AP600. This is inconsistent with the SRP. Sections 3.9.6.II.1 and 3.9.6.II.2 of the SRP state that all pumps and valves which are considered safety-related should be included in the IST program even if they are not categorized as ASME Class 1, 2, or 3. Provide a detailed discussion to justify the apparent deviation.

Furthermore, the AP600 unique designs place significant reliance on passive safety systems, but also depend on non-safety systems (which are traditional safety systems in current LWRs) to prevent challenges to passive systems. Provide information demonstrating the testability of both safety-related valves and important non-safety pumps and valves.

210.26 Section 3.9.6 of the SSAR, particularly Section 3.9.6.3, indicates that relief from the testing requirements of ASME Section XI may be requested when compliance with the Code requirements is not practical.

All plants that have been licensed to operate by the NRC have been permitted to request relief from the ASME Section XI IST rules for pumps and valves. These pumps and valves are generally installed in systems in which it is impractical to meet the Section XI rules because of limitations in the system design which preclude testing without significant design changes. In other cases, the staff granted requests for relief because imposition of the Section XI rules would have resulted in hardships to the licensee without a compensating increase in the level of safety. The underlying reason for the regulation allowing these reliefs from the Code was that the detailed system designs for all of these plants were completed prior to the time that the staff began to require the rules of Section XI of the ASME Code.

A plant such as the AP600, for which the final design is not complete, has sufficient lead time available to include provisions for this type of testing in the detailed design of applicable piping systems. Therefore, the staff concludes that a more explicit commitment that the AP600 will be designed to accommodate the applicable code requirements for IST of pumps and valves should be provided, without the expectation that requests for relief from the applicable code testing requirements will be necessary. However, with regard to subsequent or future code revisions to the applicable ASME Code for the AP600 plant, requests for relief from certain <u>updated</u> code requirements may still be submitted for staff review in accordance with 10 CFR 50.55a(g). Revise Section 3.9.6 of the SSAR to provide such a commitment.

STRUCTURAL ENGINEERING

<u>Chapter 3</u>

220.1 What is the "ASME Design Report" and when will it be available for review? Section 3.8.2.1.1 of the SSAR states that "The information contained in this subsection is based on the design specification and <u>preliminary</u> design and analysis of the vessel. Final detailed analyses will be documented in the ASME Design Report." Section 3.8.2.4.1 of the SSAR refers to a "<u>Preliminary</u> analyses ..." Justify the use of preliminary information in the application for design certification (Section 3.8.2.1.1).

220.2 The SSAR indicates that the design external pressure is 2.5 psig. How has this been considered in the analysis of the containment? What other loadings in Table 3.8.2-1 of the SSAR are to be combined with the external pressure (Section 3.8.2.1.1)?

220.3 There are no shear and tension connectors between the containment vessel and basemat and between the containment vessel and internal structures. Vertical and lateral loads on the containment vessel and internal structures are transferred to the basemat below the vessel by friction and bearing. How is the potential for relative motion between steel and concrete parts in this region under the various loading combinations considered in the design (Section 3.8.2.1.2)?

220.4 The loading combinations listed in Table 3.8.2-1 of the SSAR will induce compressive stresses into the containment shell. Describe how buckling considerations were, or will, be checked. In particular, seismic loading and localized crane loadings should be included in your discussion (Section 3.8.2.3).

220.5 Provide a more detailed description of the ultimate capacity evaluation of the cylindrical portion. Describe how strains in the vicinity of local features (such as stiffeners and penetrations) have been incorporated into the analysis. The area replacement rule may satisfy strength considerations if sufficient ductility exists, i.e., if locally high straining does not cause premature rupture. How will this be verified for the AP600 containment; i.e., what are the local strain levels at 144 psig (Section 3.8.2.4.2.1)?

220.6 Describe the analytical model used in the BOSOR-5 analysis. Justify mesh size (Section 3.8.2.4.2.2).

220.7 Does "yield" refer to surface stresses or middle surface stresses in Paragraph 2 of Section 3.8.2.4.2.2 of the SSAR?

220.8 Residual stresses are known to reduce buckling capacity. Describe how residual stresses were incorporated into the analysis (Section 3.8.2.4.2.2).

220.9 Clarify the discussion of capacity reduction factors and factors of safety in Section 3.8.2.4.2.2 of the SSAR. Capacity reduction factors are intended to reduce the theoretical buckling values to the predicted buckling strength. They account for imperfections and are usually based upon a correlation of theory and experiment. Factors of safety must be applied in addition to the capacity reduction factors. Factors of safety relate to uncertainties in loading and variability of analytical predictions.

220.10 The argument for a reduced Level C factor of safety of 1.5 in the first two bullets of Paragraph 6 of Section 3.8.2.4.2.2 of the SSAR is not clear, based on the comment in Q220.9 and the following definitions:

a. The theoretical buckling load is calculated from an analytical model which does not include imperfections.

b. The predicted buckling load represents the load at which buckling is actually expected. It includes imperfection effects. The predicted buckling load may be found as the theoretical buckling load times the capacity reduction factor.

c. The allowable buckling load is the predicted load divided by the factor of safety. The capacity reduction factor (<u>not</u> the factor of safety) is intended to include imperfection effects.

The last bullet indicates that a reduced factor of safety is permissible because of the low probability of Level C loading. However, this has already been recognized by ASME when it permits a 20 percent increase in allowable stresses for Level C over level A (Paragraph 6). Justify the use of a factor of safety of 1.5 for Level C loading.

220.11 In the case of the torispherical head, the theoretical buckling load is 176 psig. With a capacity reduction factor of 0.79, the predicted buckling load is 137 psi. With a Level C buckling factor of safety of 2.5, the Level C allowable buckling load would be 55 psig and not the 70 psig stated in Paragraph 6 of Section 3.8.2.4.2.2 of the SSAR. Clarify why no capacity reduction factor was used in the 70 psig calculation.

220.12 Paragraph 4 of Section 3.8.2.4.2.2 of the SSAR states that buckling of the head is not a consideration in the ultimate capacity of the containment because of post-buckling considerations. This argument is used again in bullet three of Paragraph 6 of the same section to justify a reduced factor of safety of 1.5 for Level C buckling. The argument is based upon the post-buckling behavior of only two tests. Provide evidence that the post-buckling strains for this head do not exceed acceptable limits. For example, what strain levels exist in the head at 144 psig, which is above the initial buckling load of 137 psig predicted in the SSAR?

220.13 The buckling factor of safety for the equipment hatch is listed in Section 3.8.2.4.2.3 of the SSAR as 1.67, following ASME N-284. What should the factor of safety be for Level C? In Section 3.8.2.4.2 of the SSAR, factors of safety of 2.5, 1.5, and 1.67 have been suggested. Justify the selection.

220.14 Demonstrate that the equipment hatch seal will not leak at the ultimate capacity. As the containment experiences large strains and displacements, there will tend to be a mismatch of the hatch shape and the cylindrical sleeve. The hatch portion of the seal will tend to displace into a circular shape whereas the cylindrical sleeve portion an elliptical shape. The two different displaced seal shapes can create a mismatch to result in seal leakage (Section 3.8.2.4.2.3).

220.15 Describe how penetrations and penetration reinforcements will be analyzed for buckling. The area-replacement rule may satisfy tensile strength requirements, but it does not necessarily satisfy buckling requirements (Section 3.8.2.4.2.5).

220.16 It is not clear what Note 3 in Tables 3.8.4-1 and 3.8.4-2 means. Does it mean that pipe will not rupture if the pipe and its supports are designed for seismic loads? Table 3.8.4-2 should include the load combination 1.2D+1.7W, in accordance with SRP 3.8.4 for other seismic Category I structures or justification should be provided for deviation from the SRP (Section 3.8.4.3.2.2).

220.17 The bearing stress of 33.6 ksf due to the dead load, live load, and safe shutdown earthquake described in Section 3.8.5.5.1 of the SSAR should be included in Table 2.0-1 as the minimum dynamic soil bearing capacity. Modify the table or provide justification for not doing so.

220.18 The equations with a square root term in Section 3A.3.1.3 of the SSAR appear incorrect. Correct or clarify them.

220.19 Provide the basis for the factors used in defining allowable stresses for the loading conditions disussed in Section 3A.3.1.3 on p. P3A-3.

220.20 Provide a detailed description and demonstrate the adequacy of the mechanical connections used to join a module with reinforcing bars in the concrete (Section 3A.5).

SEISMIC DESIGN

Chapter 3

230.1 Section 3.7 of the SSAR states that Non-Category I facilities are designed in accordance with the Uniform Building Code (UBC), Zone 2A, requirements. Clarify the intent of this statement which implies that any sites in Zones 2B, 3, and 4 with more severe seismic requirements are excluded from the standard design. Note that this requirement will exclude a large part of the western United States from site selection.

230.2 Clarify the text in Section 3.7 of the SSAR regarding whether the Non-Category I facilities include seismic Category II structures, such as the Turbine Building, Annex Buildings I and II, and Solid Radwaste Building.

230.3 Section 3.7 of the SSAR states that the operating basis earthquake (OBE) has been eliminated as a design requirement for the AP600. Regulatory Guide 1.143 allows radwaste buildings to be designed to the OBE. Regulatory Guide 1.27 allows certain parts of the ultimate heat sink to be designed for the OBE. What is the AP600's seismic design basis for these facilities?

230.4 Section 3.7 of the SSAR states that the operating basis earthquake (OBE) has been eliminated as a design requirement for the AP600. Appendix A of 10 CFR Part 100 requires that if vibratory ground motion exceeding the OBE occurs, shutdown of the plant will be required. State what the AP600 excitation level is specified for plant shutdown purposes.

230.5 Section 3.7 of the SSAR states that the cumulative absolute velocity (CAV) approach according to EPRI Report NP-5930 will be used for plant shutdown criteria following an earthquake. The CAV calculation discussed in EPRI NP-5930 has been amended. The standardized CAV calculation is discussed in EPRI Report TR-100082. The guidelines for nuclear plant response to an earthquake are discussed in EPRI Report NP-6695. State in greater detail what the AP600 plant procedures are following an earthquake occurrence.

230.6 Section 3.7.1.2 of the SSAR states that the "TAFT" earthquake time history was used to generate synthetic time histories for AP600 seismic design. The SSAR presents spectrum comparison between the AP600 damped seismic design response spectra and the corresponding RG 1.60 response spectra anchored to 0.3 g for the damping ratios of 2, 3, 4, and 7% in Figures 3.7.1-6 through 3.7.1-8. However, the SSAR should also provide a spectrum comparison for the case with a damping ratio of 5%. Provide such a spectrum.

230.7 Provide the basis of the damping values for cable trays, conduits, and their supports presented in Table 3.7.1-1 and Figure 3.7.1-13 (Section 3.7.1.3).

230.8 Provide a description and its technical basis for the "strain energy method" used to model composite damping (Section 3.7.1.3).

230.9 ASCE Standard 4-86, "Seismic Analysis of Safety-Related Nuclear Structures and Commentary," which has not been endorsed by the staff, should be submitted and docketed for the staff review for the AP600 standard design (Section 3.7.1.3).

230.10 Three design soil profiles, which includes a hard rock site, are selected in the seismic analysis of Category I structures. Demonstrate that this set of seismic analyses will provide conservative design envelopes for all potential sites or confirm that site-specific seismic analyses will be performed for a selected site (Section 3.7.1.4).

230.11 Section 3.7.2 of the SSAR provides a very general design requirement for Category II structures by stating that "Seismic Category II building structures are designed and/or physically arranged so that the safe shutdown earthquake (SSE) could not cause unacceptable structural interaction with or failure of their adjacent seismic Category I structures, systems, and components." Provide detailed analysis methods and design criteria that will be used to meet this general design requirement. For example, what seismic analysis will be performed for Category II structures?

230.12 The AP600 standard design employs modular construction for the containment structural internals. Appendix 3A of the SSAR discusses design and analysis procedures for sizing up structural members and preparing fabrication details. The staff notes that, in Section 3.7.2 of the SSAR, the AP600 seismic systems analysis is described for conventional approaches to seismic Category I structures, which may not be totally applicable to structures comprised of the modular units used in the AP600 design. Revised Section 3.7.2 to include a discussion on seismic behavior and the corresponding design analysis methods for the AP600 modular constructions.

230.13 Section 3.7.2.1.1 of the SSAR states that response spectrum analyses are performed only for the hard rock site and the hard rock site condition governs the seismic response forces and moments for the seismic Category I building structures. Was the case with the containment vessel founded on the hard rock site also analyzed with the response spectrum method? If not, describe the analysis method and demonstrate its adequacy for analyzing the soil-containment vessel interaction system.

230.14 How are the truss elements used in the stick model of Figure 3.7.2-4 (Section 3.7.2.1.2)?

230.15 Section 3.7.2.1.2 of the SSAR states that "Certain subsystems...are analyzed using the time histories obtained from a series of soil-specific analyses." What are these soil-specific analyses? Provide details of these analyses.

230.16 Describe the method used to construct a stick model from the axisymmetric shell model of the containment vessel (Section 3.7.2.3.2).

230.17 Describe the procedures used to consider effects of adjacent structures (Turbine, Annex I, Annex II, and Solid Radwaste Buildings) on the SSI analysis of seismic Category I structures (Section 3.7.2.4).

230.18 Provide the basis for the statement, "The selected soil conditions envelop the potential variation of soil properties,..." See comments on Section 3.7.1.4 (Q230.10) (Section 3.7.2.4).

230.19 How is the "enveloped floor response spectra" defined? Will they bound the floor response spectra obtained from the three design soil profiles? Figure 3.7.2-27 of the SSAR which shows spectrum broadening appears to suggest that only a single floor response spectrum is involved and does not reflect the enveloping process described in the last paragraph of Section 3.7.2.5 of the SSAR.

230.20 Provide the basis for the third method of Section 3.7.2.6 of the SSAR for combining the results from analyzing three components of earthquake motion. In the time-history analyses, were all three methods used interchangeably to generate a single set of results such as floor response spectra for all locations of the seismic Category I structures?

230.21 The modal responses of the response spectrum analysis of structures are combined using the square root of the sum of squares (SRSS) method. The SRSS method is in agreement with RG 1.92 if no closely spaced modes are present. Describe the method used for the cases with closely spaced modes (Section 3.7.2.7).

230.22 The last sentence of Section 3.7.2.8 of the SSAR states that "These structures are analyzed and designed to prevent their failure under the SSE." Provide detailed analysis methods and design criteria that will use to prevent their failure under the SSE. See staff comments on Section 3.7.2 above.

230.23 Section 3.7.4.2 of the SSAR indicates that four triaxial acceleration sensors will be installed at an AP600 plant. Regulatory Guide 1.12 "Instrumentation for Earthquakes" is presently being revised by the NRC staff. The draft guide calls for 7 or 8 triaxial acceleration sensors at various locations within the plant site. Discuss in detail the AP600 position with respect to amending the SSAR to comply with the RG 1.12 revision.

SITE CHARACTERISTICS

Chapter 2

231.1 Clarify the following statement in Section 2.5 (p. 2.0-3): "For the site where the soil characteristics differ significantly...site-specific soil structure interaction analyses may be performed to demonstrate acceptability..." Referring to Section 2A.6 of the SSAR in which the base rock depth of design soil profiles was specified at 37 m (120 ft), will site-specific seismic analyses be required if the site base rock depth is, for example, 46 m (150 ft), which is deeper than the 37 m (120 ft) condition analyzed? See staff comment on Section 3.7.1.4 of the SSAR (Q230.10).

231.2 Provide the floor response spectra at the four locations referenced as the basis for demonstrating that the site seismic conditions are within the AP600 design basis. This should be documented in the SSAR (Section 2.5).

231.3 Section 2.5 of the SSAR states that, for sites where soil characteristics differ significantly from those used in the generic sensitivity analysis, the COL applicant may perform site-specific soil structure interaction analysis and compare the site-specific floor response spectra at four locations in the superstructure. Explain why a comparison of the ground response spectra at the foundation level will not be made to demonstrate that the site-specific seismic conditions are within the AP600 design basis.

231.4 Table 2.0-1 of the SSAR requires minimum soil bearing strength to be 575 kPa (12 ksf). Provide the basis for accepting a bearing stress of 1610 kPa (33.6 ksf) in Section 3.8.5.5.1 (Section 2.5).

231.5 The AP600 design assumes an upper bound value of 2,440 m/s (8,000 fps) for the shear wave velocity of the hard rock site. Shear wave velocities of 3,050 to 3,350 m/s (10,000 to 11,000 fps) are not uncommon for hard rock in the eastern United States. Soil structure interaction (SSI) studies indicate that SSI effects are present for rocks with shear wave velocities up to at least 3,050 m/s (10,000 fps). Also, an important consideration in the amplification of the ground motion is the ratio of the shear wave velocity of the overlying soil to that of the rock. Provide in detail the SSI analyses that formed the basis of the conclusions in the SSAR and elaborate on corresponding findings that soil amplification at soil sites does not significantly alter the input motions for the seismic analysis of the systems and subsystems (Section 2.5).

231.6 Explain why the item, "lateral earth pressure loads" in Table 1.8-1 of the SSAR is not an item to be addressed by the combined license applicant (Section 2.5).

231.7 State the reasons for not including a discussion in the SSAR of the analysis procedures that would be used for evaluating the stability of slopes, dams, and embankments (Section 2.5).

231.8 Certain soils may liquefy under vibratory ground motion. What level and duration of ground motion is used to assess the soil liquefaction potential for the AP600 (Section 2.5)?

231.9 On External Events Analyses, Seismic Margin Assessment, Appendix H, a review level earthquake of 0.45 g was identified for the seismic margin assessment to demonstrate sufficient margin over the SSE of 0.30 g. The purpose of the seismic margins analysis is to test the plant's vulnerability to severe accidents beyond the design basis. Seismic margins studies and seismic probabilistic risk assessments conducted for operating nuclear power plants have shown the plant HCLPF to be 2 to 3 times the design value. In view of this, explain why the SSAR chose such a low value for review level earthquake of 0.45 g, which is only 1.5 times the SSE of 0.3 g?

Appendix 2A

231.10 Does the AP600 design specify the control ground motion at an actual or hypothetical rock outcrop for sites with one or more thin soil layers overlying a rock, as specified in Section 3.7.1 of the SRP. If it does, where is this discussed in the SSAR? If it does not, justify why the SSAR does not follow the SRP (Section 2A.3).

231.11 The SSAR states that the "customary" +50% or -50% variation in low strain shear modulus (G_{max}) for each profile was not applied in the free field analysis because the generic soil profiles considered include a wide range of shear wave elocities. Explain how the AP600 design satisfies the provisions of Section 3.7.2 of the SRP which specifiy variation of G_{max} by a factor of 2, i.e., +2 G_{max} and -0.5 G_{max} (Section 2A.4).

231.12 Section 2A.5 of the SSAR states that "To identify the governing site properties and profiles...two-dimensional SSI analyses were performed..." The use of the word "governing" suggests that the AP600 may not seek for a "bounding" standard seismic design. As such, clarify what the site-specific analysis procedures and criteria are in addition to those site parameters comparison requirements of Section 2.5 of the SSAR.

231.13 Provide the basis for the following statement in Section 2A.5 of the SSAR: "The results from the horizontal and vertical analyses were not combined..." This statement does not agree with the staff's position delineated in RG 1.92. Would the foundation rocking contribute significantly to the translational seismic response at higher elevation of the containment vessel and the shield building (including the containment cooling system water tank)?

231.14 Clarify the following statement in Section 2A.6 of the SSAR: "Based on the site interface requirements...it is concluded that enveloped responses for the design soil profiles adequately envelop the responses of the AP600 plant structures for ...shear wave velocity greater than or equal to 305 m (1000 ft) per second." How are the "enveloped responses" defined? Will the envelopes of the foundation-level response spectra resulting from the SSI analyses of the three design soil profiles be used for the seismic Category I structures and floor response spectra for the subsystems?

INSERVICE INSPECTION

Chapter 3

Inservice Inspection of the Containment

250.1 Section 3.8.2.7 of the SSAR indicates that the inservice inspection of the containment vessel will be in accordance with Subsection IWE of the 1989 edition of the ASME Section XI Code. However, Subsection IWE has been revised recently to incorporate operating experience. Therefore, provide information to indicate that the Section XI requirements are to be augmented with the requirements of Subsection IWE, as revised.

250.2 Discuss Westinghouse's proposed procedures in applying the revised Subsection IWE of the ASME Section XI Code to identify locations in the containment vessel with propensity for corrosion (Section 3.8.2).

Chapter 5

Inservice Inspection of Class 1 Components

250.3 Demonstrate that all ASME Code Class 1 components will be designed and be provided with access to enable the performance of ASME Section XI inspections in the installed conditions as required by 10 CFR 50.55a(g). Because the RCPB components will be designed to the 1989 edition, 1989 addenda, of the ASME Code as described in Section 5.2.1.1 of the SSAR, demonstrate that adequate design and access provisions will be incorporated to permit inspection for those components that are required to be inspected by the 1989 edition, 1989 addenda, of the ASME Section XI Code (Section 5.2.4).

250.4 Demonstrate that the preservice inspection (PSI) of all ASME Code Class 1 components will meet the 1989 edition, 1989 addenda, of the ASME Section XI Code as required by 10 CFR 50.55a(g). Because the PSI requirements have been established, 10 CFR 50.55a(g) does not have provisions for relief requests for impractical PSI examination requirements. Provide information to confirm that all PSI requirements will be met (Section 5.2.4).

250.5 ASME Section XI indicates that the PSI should be conducted with equipment and techniques equivalent to those that are expected to be used for subsequent inservice inspection (ISI). The PSI provides the baseline information for reference in subsequent ISI. For example, if the ISI of piping weld is expected to be performed with ultrasonic techniques, the PSI should also be based on ultrasonic techniques. Provide information to confirm that this requirement will be satisfied for all ASME Code Class 1 components (Section 5.2.4).

250.6 Provide information to confirm that Article IWA-1500, "Accessibility," of Section XI of the ASME Code will be satisfied for all ASME Code Class 1 components (Section 5.2.4).

250.7 The ASME has published Appendix VII, "Qualification of Nondestructive Examination Personnel for Ultrasonic Examination," and Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," in Section XI (Division 1) of the ASME Code. The NRC has published (in the Federal Register) its intent to reference in 10 CFR 50.55a(b) the ASME Section XI edition that includes the published Appendix VII. In addition, the NRC staff has established a technical contact to coordinate the implementation of Appendix VIII. Therefore, indicate that Section XI requirements are to be augmented with the requirements in Appendices VII and VIII for all ASME Code Class 1 components (Section 5.2.4).

250.8 ASME Code Class 1, 2, and 3 carbon and low-alloy steel piping items that are susceptible to wall thinning as a result of the single-phase (water) erosion/corrosion phenomenon will be subject to examination in accordance with Subsection IWH of ASME Section XI. Therefore, indicate that Section XI requirements are to be augmented with the requirements of Subsection IWH for all ASME Code Class 1 components (Section 5.2.4).

250.9 Table IWB-2500-1 in Section XI of the ASME Code requires the examination of Class 1 piping welds, with a Section III fatigue cumulative usage factor (CUF) exceeding 0.4, at every inspection interval. Confirm that the value of CUF to be used will correspond to the projected 60-year plant design life (Section 5.2.4).

Steam Generator Tube Inservice Inspection

250.10 Describe the steam generator tube inservice inspection program, such as the inspection technique, provisions for the selection and sampling of tubes, the inspection intervals, the actions to be taken in the event defects are identified, and reporting requirements (Section 5.4.2).

250.11 Figure 5.4-2 in the SSAR does not show the orientation and location of all of the access points in the steam generator. Provide drawings to show the secondary side access points in the steam generator.

250.12 Discuss whether the four 15 cm (6 in) handholes located just above the tubesheet are of sufficient size to allow for effective sludge lancing, retrieval of loose parts, and/or inspection of the tube bundle by portable inspection equipment (e.g. video equipment) (Section 5.4.2).

250.13 Describe the design provisions for tube indexing for facilitation of tube identification and location during inservice inspections (Section 5.4.2).

250.14 Describe the physical location of the internal deck plates used to gain access to the U-bend area. Clarify the statement in Section 5.4.2.5 of the SSAR that "for proper functioning of the steam generator, some of the deck-plate openings are covered with welded but removable, hatch plates."

250.15 Describe the features incorporated in the design that enhance inspection of the steam generator tubes without manned entry. Discuss whether the design features support the use of current robotic equipment used in steam generator tube inspection and repair. In addition, discuss whether verification have been performed, by computer simulation and/or mockup, to ensure that the design will facilitate not only the use of robotic manipulators in inspecting all of the tubes within the steam generator but also in inserting the robotics into the steam generator (Section 5.4.2).

250.16 When in the fabrication procedure will the shop examination of the tubing be performed? Describe the procedures and precautions taken to ensure the integrity of the tubes during final assembly, shipment, and installation of the steam generators (Section 5.4.2).

250.17 Provide clarification on what it considers "more capable equipment" or a "suitable eddy current inspection system" as compared with the equipment described in paragraph C.2.c of Regulatory Guide 1.83 (Section 5.4.2).

250.18 Describe the corrective measures that will be implemented to disposition leaking tubes, defective tubes, and tubes with imperfections exceeding the plugging limits (Section 5.4.2).

250.19 Provide clarification to exceptions to criteria C.2.a.(2) and C.2.a.(4) of Regulatory Guide 1.121. In particular, describe how the proposed change will affect the margin of safety currently observed. Describe the statistical analysis of the tensile test data that is used in the development of the expected material strength properties. Also discuss whether the calculation of the tube minimum wall requirements will be based on the lowest values for the material properties, i.e., the lowest values from statistical analyses or from the ASME Code (Section 5.4.2).

250.20 Provide technical justifications for exceptions to criteria C.2.a.(5)-(6) and C.2.b of Regulatory Guide 1.121 (Section 5.4.2).

250.21 Where will the provisions for inservice inspection of steam generator tubes be implemented, e.g., plant technical specifications (Section 5.4.2)?

Chapter 6

Inservice Inspection of Class 2 and 3 Components

250.22 Demonstrate that all ASME Code Class 2 and 3 components will be designed and be provided with access to enable the performance of ASME Section XI inspections in the installed conditions as required by 10 CFR 50.55a(g). Further, confirm that Class 2 and 3 components will be designed to the 1989 edition, 1989 addenda, of the ASME Code. Verify that the applicable inspections are those in the 1989 edition, 1989 addenda, of the ASME Section XI Code (Section 6.6).

250.23 Demonstrate that the preservice inspection (PSI) of all ASME Code Class 2 and 3 components will meet the 1989 edition, 1989 addenda, of the ASME Section XI Code as required by 10 CFR 50.55a(g). Because the PSI requirements have been established, 10 CFR 50.55a(g) does not have provisions for relief requests for impractical PSI examination requirements. Provide information to confirm that all PSI requirements will be met (Section 6.6).

250.24 ASME Section XI indicates that the PSI should be conducted with equipment and techniques equivalent to those that are expected to be used for subsequent inservice inspection (ISI). The PSI provides the baseline information for reference in subsequent ISI. For example, if the ISI of piping weld is expected to be performed with ultrasonic techniques, the PSI should also be based on ultrasonic techniques. Provide information to confirm that this requirement will be satisfied for all ASME Code Class 2 and 3 components (Section 6.6).

250.25 Provide information to confirm that Article IWA-1500, "Accessibility," of Section XI of the ASME Code will be satisfied for all ASME Code Class 2 and 3 components (Section 6.6).

250.26 The ASME has published Appendix VII, "Qualification of Nondestructive Examination Personnel for Ultrasonic Examination," and Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," in ASME Section XI (Division 1). The NRC has published in the Federal Register its intent to reference in 10 CFR 50.55a(b) the ASME Section XI edition that includes the published Appendix VII. In addition, the NRC staff has established a technical contact to coordinate the implementation of Appendix VIII. Therefore, provide information to indicate that Section XI requirements are to be augmented with the requirements in Appendices VII and VIII for all ASME Code Class 2 and 3 components (Section 6.6).

250.27 ASME Code Class 1, 2, and 3 carbon and low-alloy steel piping items that are susceptible to wall thinning as a result of the single-phase (water) erosion/corrosion phenomenon will be subject to examination in accordance with Subsection IWH of ASME Section XI. Therefore, provide information to indicate that Section XI requirements are to be augmented with the requirements of Subsection IWH for all ASME Code Class 2 and 3 components (Section 6.6).

Chapter 10

Turbine Disk Integrity

250.28 If drilled holes will be present in the rotor, discuss the preservice inspection requirements for them (Section 10.2.3).

250.29 Provide information to confirm that the inservice inspection program discussed in Section 10.2.3.6 of the SSAR will ensure that the failure and missile generation probability will be less than 10^{-4} per year. (See Q251.1)

COMPONENT INTEGRITY

Chapter 3

251.1 The staff's position regarding turbine maintenance and inspection is that the turbine maintenance and inspection program be implemented to ensure that the failure and missile generation probability is less than 10^{-4} per year for a favorably oriented turbine system [see letter from C. E. Rossi (NRC) to J. A. Martin (Westinghouse) dated February 2, 1987]. Describe how this position will be met (Section 3.5.1.3).

Chapter 5

Pump Flywheel Integrity

251.2 Westinghouse proposes to use a depleted uranium alloy casting in an Inconel alloy welded enclosure to construct the pump flywheel. These materials are not addressed in Section 5.4.1.1 of the SRP and Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity." Provide technical justifications for the use of these materials (Section 5.4.1).

251.3 Westinghouse indicates that the fracture toughness guidelines in Section 5.4.1.1 of the SRP and Regulatory Guide 1.14 are not applicable to depleted uranium alloy castings. Provide information on the fracture toughness properties for this material and propose fracture toughness requirements with technical justifications (Section 5.4.1).

251.4 Provide information on the fabrication process and resulting quality for the depleted uranium alloy casting (Section 5.4.1).

251.5 Section 1A of the SSAR indicates that the AP600 design meets the guidelines of Regulatory Position 1.d in Regulatory Guide 1.14. However, the flywheel, including the enclosure welds, will not be inspected. Discuss how the flywheel design meets Regulatory Position 1.d.

251.6 Regulatory Positions 2.c, 2.d, and 2.e in Regulatory Guide 1.14 recommends that an analysis be submitted for staff review. Provide the analysis with appropriate technical justifications. Further, because no inservice inspection for the flywheel is being proposed, describe the flaw size assumed in its analysis (Section 5.4.1).

251.7 Section 1A of the SSAR indicates conformance with Regulatory Position 2.f in Regulatory Guide 1.14. Provide information to support this statement.

251.8 Section 1A of the SSAR indicates conformance with Regulatory Position 2.g in Regulatory Guide 1 14, relating to the flywheel overspeed due to a postulated pipe rupture. Section 5.4.1.3.6.3 of the SSAR appears to assume the application of leak-before-break (LBB) for all high-energy piping 10 cm (4 in) in diameter or larger. Since the outcome of the staff's review of the application of LBB to the AP600 design is uncertain, the staff recommends that Westinghouse discuss how the flywheel conforms with RG 1.14 if the criteria of Section 3.6.2 and BTP MEB 3-1 is used to determine pipe break size.

251.9 Section 1A of the SSAR indicates that Westinghouse is taking exception to Regulatory Position 4.a in Regulatory Guide 1.14. Propose an alternative to this position with appropriate technical justifications.

251.10 Performance of inservice inspection of the flywheel should be considered. If the ISI procedures in Section 5.4.1.1 of the SRP is not applicable to uranium flywheels, propose alternative inservice inspection procedures with appropriate technical justifications (Section 5.4.1).

251.11 Section 1A of the SSAR states that a flywheel rupture will be contained within the stator shell. Provide an analysis and technical justifications supporting this statement.

251.12 Section 1A of the SSAR indicates that a "small" flywheel rupture or leak in the enclosure will not result in stresses in the pressure boundary to cause a break. Provide information to clarify what is the intent of the term "small" flywheel rupture. The staff is concerned with the rupture of the flywheel into large fragments of high energy.

251.13 Section 5.4.1.3.6.3 of the SSAR indicates that ultrasonic inspection of the uranium following final machining will be based on ASTM A388 as modified for uranium. Identify any modifications to the application of ASTM A388 to the AP600 design with appropriate technical justifications. In addition, demonstrate that this preservice inspection is equivalent to that in Section III of the ASME Code.

251.14 Demonstrate that the construction of the flywheel enclosure meets Section III of the ASME Code, including inspection (Section 5.4.1).

251.15 Demonstrate that the design overspeed of the flywheel is at least 10% above the highest anticipated overspeed (Section 5.4.1).

251.16 Show that the combined stresses for the uranium flywheel at the normal operating speed, due to centrifugal forces and the interference fit of the wheel on the shaft, is less than 1/3 of the minimum specified yield strength (Section 5.4.1).

251.17 Discuss how the limit in Q251.16 is met for the flywheel enclosure and associated welds (Section 5.4.1).

251.18 Show that the combined stresses for the uranium flywheel at the design overspeed, due to centrifugal forces and the interference fit, is less than 2/3 of the minimum specified yield strength (Section 5.4.1).

251.19 Discuss how the limit in Q251.18 is met for the flywheel enclosure and associated welds (Section 5.4.1).

251.20 Demonstrate that the shaft and the bearings supporting the flywheel will be able to withstand any combination of loads from normal operation, anticipated transients, the design basis of loss-of-coolant accident, and the safe shutdown earthquake (Section 5.4.1).

251.21 Identify the materials for the flywheel enclosure and associated welds. Provide technical justifications to show that the flywheel enclosure and associated welds are resistant to stress corrosion cracking, especially if Inconel 600 or 182 materials will be used (Section 5.4.1).

251.22 Demonstrate that the uranium flywheel is resistant to stress corrosion cracking or other potential degradation mechanism in a reactor coolant environment (Section 5.4.1).

251.23 Table 5.4-2 in the SSAR lists the flywheel material specifications. Provide the technical basis for these specifications.

Chapter 10

Turbine Disk Integrity

251.24 Section 10.2.3.2 of the SSAR indicates that flaws may be acceptable in the rotor if the flaws can be shown not to grow to critical sizes. A flaw growth evaluation to demonstrate structural integrity in lieu of flaw removal is not consistent with the ASME Section III Code which does not permit a flaw evaluation. Discuss how the acceptance criteria in Section III and Section V of the ASME Code are met.

251.25 Demonstrate that the fracture appearance transition temperature (50% FATT) as obtained from Charpy tests performed in accordance with ASTM A370 will be no higher than -18°C (0°F) for low-pressure turbine rotors (Section 10.2.3).

251.26 Provide information to show that the Charpy V-notch energy at the minimum operating temperature of low-pressure rotors in the tangential direction will be at least 82 J (60 ft-lb) (Section 10.2.3).

251.27 Provide information in Section 10.2.3.2 of the SSAR to demonstrate that the ratio of the fracture toughness "K_{Ic}" of the rotor material to the maximum tangential stress at speeds from normal to design overspeed will be at least 3.2 $\sqrt{cm}$ (2 $\sqrt{in}$), at minimum operating temperature.

251.28 Provide information to show that sufficient warmup time will be specified in the turbine operating instructions to ensure that toughness will be adequate to prevent brittle fracture during startup (Section 10.2.3).

251.29 Section 10.2.3.2 of the SSAR indicates that fracture toughness properties will be obtained using procedures more conservative than those in Scientific Paper 71-1E7-MSLRF-P1, J. A. Begley and W. A. Logsdon, Westinghouse Electric Corp., July 26, 1971. Provide the paper and additional information to demonstrate this conservatism. In addition, verify that Acceptance Criterion II.2.c in Section 10.2.3 of the SRP will be met.

251.30 Section 10.2.3.4 of the SSAR indicates that the low-pressure turbine element has a central bore while the high-pressure turbine element does not. The staff considers a central bore desirable to remove impurity inclusions from integral rotors. Provide technical justifications for not boring the high-pressure rotor.

251.31 Confirm that each finished rotor will be subjected to 100% volumetric (ultrasonic), surface, ard visual examinations using procedures and acceptance criteria equivalent to those specified for Class 1 components in Sections III and V of the ASME Code (Section 10.2.3).

251.32 Discuss conformance with guidance in Acceptance Criteria II.4.a, II.4.b, II.4.c, II.4.d, and II.4.e in Section 10.2.3 of the SRP (Section 10.2.3).

MATERIALS ENGINEERING

Chapter 3

252.1 Discuss the considerations given to the quenching of the containment in the event of a severe accident.

Leak-Before-Break

252.2 Section 3.6.3 of the SSAR indicates that the leak-before-break (LBB) methodology will be used to eliminate the dynamic effects of postulated pipe ruptures from the design basis. The SSAR indicates that the scope of LBB application is high-energy ASME Code Section III Class 1, 2, and 3 piping of 10 cm (4 in) in nominal diameter or larger. Identify specific piping being considered for LBB applications (see Q210.6).

252.3 Perform bounding LBB analyses for each of the LBB candidate piping, including evaluations for susceptibility to potential degradation mechanisms for the projected 60-year plant design life. Provide the analyses (Section 3.6.3).

252.4 Describe the procedures to be used by the COL applicant to verify that the actual material properties and final, as-built piping analyses are within the limits in the bounding LBB analyses (Section 3.6.3).

252.5 Section 3.6.3 of the SSAR indicates that Class 2 and 3 piping are within the LBB scope. The staff has not approved the application of LBB for these piping for operating reactors. There are differences in ASME Code requirements between Class 1 and Class 2 and 3 piping. Discuss the significance of these differences on ensuring piping structural integrity and describe procedures to address them.

For example, the ASME Code does not require a fatigue analysis for Class 2 and 3 piping. [illegible]uss how the fatigue resistance of the LBB Class 2 and 3 piping will be addressed. As another example, the inservice inspection requirements for Class 2 piping is based on a sampling basis and Class 3 piping is based on visual inspections. Discuss any augmented inservice inspection for Class 2 and 3 LBB piping.

252.6 Section 3.6.3 of the SSAR indicates that LBB may be applied for portions of piping outside containment. Provide information to demonstrate the reliability, effectiveness, sensitivity, and timeliness of leakage detection methods and procedures selected for outside containment.

252.7 Section 16.1 of the SSAR indicates that the AP600 technical specifications limit the unidentified reactor coolant system leakage to less than 0.03 L/s (0.5 gpm). This leakage limit is used in an LBB analysis for piping inside the containment. Describe administrative controls to ensure that any increase in the unidentified leakage limit in the AP600 technical specifications will initiate a reevaluation of the LBB analyses (Section 3.6.3).

252.8 Demonstrate the reliability, effectiveness, sensitivity, and timeliness of leakage detection methods and procedures selected for inside containment to detect a 0.03 L/s (0.5 gpm) unidentified leakage (Section 3.6.3).

252.9 The standard design employs modular construction for various types of components of the containment structural internals. Appendix 3A of the SSAR discusses design and analysis procedures for sizing up strurtural members and preparing fabrication details. Although seismic design analysis approaches for systems and subsystems are presented in Sections 3.7.2 and 3.7.3, which generally address conventional Seismic Category I structures, they may not be totally applicable to structures comprised of the modular units used in the AP600 design. Discuss seismic behavior and the corresponding design analysis methods for the AP600 modular constructions.

252.10 Section 3B of th SSAR discusses the LBB evaluation for the reactor coolant loop piping. The SSAR indicates that two different soil conditions have been considered in deriving piping stresses. Discuss how these piping stresses represent the worst condition of all potential sites within the scope of AP600 applications.

252.11 Tables 3B-3 and 3B-4 of the SSAR give stresses used in the LBB evaluation of the reactor coolant loop piping. Provide information to clarify whether the stresses are from the stress analysis of routed or unrouted reactor coolant loop piping.

252.12 Section 3.6.3.3 of the SSAR indicates that "part through-wall flaws" may be considered at the critical locations. This is not consistent with the requirements of a LBB analysis. Provide information to clarify this statement.

252.13 Section 3.6.3 of the SSAR discusses feedwater and steam piping. The staff has not approved the application of LBB for these piping for power reactors. Provide additional discussion relating to potential susceptibility of feedwater and steam piping to degradation mechanisms, such as water/steam hammer and erosion/corrosion.

252.14 The pressurizer surge line is potentially susceptible to thermal stratification. If the surge line is within the LBB scope, describe the ASME Section III fatigue "cumulative usage factor" for the surge line for the projected 60-year plant design life and the considerations given to the thermal stratification loads in the LBB analysis (Section 3.6.3).

Reactor Coolant Piping

252.15 Section 3B.2.2 of the SSAR indicates that the reactor coolant loop piping will be fabricated from SA376 TP316LN austenitic stainless steel. The staff is not aware of the application of Type 316LN in light-water reactors. Provide operating experience and test data to demonstrate that Type 316LN is not susceptible to stress corrosion cracking in a PWR environment for the projected 60-year plant design life.

252.16 Section 3B.2.2 of the SSAR indicates that the carbon content of the austenitic stainless steel in the reactor coolant loop piping is limited to 0.035%. The staff has recommended limiting the carbon content to less than 0.02% to resist intergranular stress corrosion cracking in a BWR environment (NUREG-0313, Rev. 2, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," January 1988). Provide test data to demonstrate that austenitic stainless steel with a carbon content of 0.035% is not susceptible to intergranular stress corrosion cracking in a PWR environment for the projected 60-year plant design life.

252.17 Section 3B.2.2 of the SSAR indicates that the material used for buttering nozzles at the stainless-to-carbon steel safe ends is a high nickel alloy. Identify the specific buttering material and provide the technical basis to demonstrate this buttering material is not susceptible to stress corrosion cracking.

252.18 Carbon steel materials may be susceptible to the mechanism of dynamic strain aging which reduces the material fracture properties.[1] Describe the procedures that address the effects of dynamic strain aging.

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[1]C. W. Marschall, M. P. Landow, and G. M. Wilkowski, "Effect of Dynamic Strain Aging on Fracture Resistance of Carbon Steels Operating at Light-Water-Reactor Temperatures," ASTM STP 1074, American Society for Testing and Materials, Philadelphia, PA, 1990, pp. 339-360.

252.19 Recent fatigue test data indicates that the effects of the environment could significantly reduce the fatigue resistance of materials.[2] The specific concern relates to the reactor water and temperature environment and its synergistic interactions with strain rate. The recent data indicate that the design fatigue curves in ASME Section III Class 1 requirements may not be as conservative as originally intended. Describe the procedures that explicitly account for the effects of the environment in the fatigue analysis of components. (Cladding on base metal is not a structural material and should not be considered adequate to isolate the base metal from the effects of the environment. This is because the cladding may be breached exposing the base metal to the water environment. Further, the cladding does not insulate the base metal from the reactor temperature.) (Section 3.6.3).

Fracture Prevention of Containment Pressure Boundary

252.20 Section 3.8.2.6 of the SSAR indicates that the containment vessel materials will be impact tested according to Article NE-2000 of the ASME Code. However, Section 6.2.7 of the SRP recommends that the fracture toughness of the reactor containment pressure boundary materials should meet the fracture toughness requirements in Subsection NC of the ASME Section III Code. Provide technical justifications for this deviation.

252.21 Section 3.8.2.6 of the SSAR indicates that the containment vessel is coated to a level just below the concrete. Provide technical justifications for not coating the portion of the containment vessel that is embedded in concrete.

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[2]1. K. Iida, J. Fukakura, M. Higuchi, H. Kobayashi, S. Miyazono, and M. Nakao, "Survey of Fatigue Strength Data of Nuclear Structural Materials in Japan," Abstract of DBA Committee Report, 1988. (Presented to the American Society of Mechanical Engineers, Subgroup on Fatigue Strength, on December 5, 1988, in New York City, NY.) (Enclosure in letter, from J. Craig (NRC) to E. Griffing (Nuclear Management and Resources Council) dated July 2, 1991.)

2. M. Higuchi and K. Iida, "Fatigue Strength Correction Factors of Carbon and Low-Alloy Steels in Oxygen-Containing High-Temperature Water," Nuclear Engineering and Design, Volume 129, 1991, pp. 293-306.

3. J. B. Terrell, "Effect of Cyclic Frequency on the Fatigue Life of ASME SA-106-C Piping Steel in PWR Environments," Journal of Materials Engineering, Volume 10, Number 3, 1988, pp. 193-203.

252.22 Discuss the need for cathodic protection of the containment vessel to protect from ground water corrosion and stray current corrosion. Also, describe considerations for the location and type of cathodic protection anodes, i.e., deep bed versus mat-type anodes (Section 3.8.2).

252.23 Article NE-3121 of the ASME Code requires the consideration of corrosion in the design of the containment. Specifically, the containment thickness is to be increased over that determined by the design formulas in the ASME Code to account for corrosion. Provide a corrosion allowance for the projected 60-year plant design life with the associated technical basis (Section 3.8.2).

252.24 Demonstrate that the containment vessel is designed and provided with access to permit the performance of inspection, maintenance, and repair of all exterior and interior surfaces of the containment vessel, except for the portion embedded in concrete (Section 3.8.2).

252.25 The exterior surface of the containment vessel may be exposed to weather conditions. Discuss the effects of weather on the corrosion of the exterior surface of the containment vessel (Section 3.8.2).

252.26 Discuss the potential for corrosion of the containment vessel within the middle annulus area of the shield building, i.e., the area bounded above by a seal and below by concrete. For example, trapped moisture or fluid may cause accelerated corrosion of the containment vessel (Section 3.8.2).

252.27 Discuss the potential effects of corrosion on the reactor vessel containment due to a leak in the passive containment cooling system water storage tank atop the shield building (Section 3.8.2).

252.28 Discuss the effects of corrosion on the heat transfer capability of the containment vessel during natural circulation. Discuss acceptance criteria for the surface condition of the containment vessel in order to maintain an acceptable heat transfer capability. Also, discuss required actions when the acceptance criteria are not satisfied (Section 3.8.2).

Chapter 4

Control Rod Drive System Materials

252.29 Provide information to confirm that the materials selected for the control rod drive mechanism components exposed to the reactor coolant will meet Section III of the ASME Code (Section 4.5.1).

252.30 Identify where the application of Inconel 600 and 182 materials will be applied. Operating experience indicates that these materials are susceptible to cracking. If these materials will be used, the applicant should provide a technical discussion on their ability to resist cracking for the projected 60-year plant design life (Section 4.5.1).

252.31 Section 4.5.1 of the SSAR indicates that cobalt based alloys will be used in the control rod drive system. Activation of cobalt is a concern relating to the radioactivity in current nuclear plants. Therefore, cobalt application should be avoided in AP600 for ALARA considerations. The use of cobalt based alloys should be avoided except in cases where no alternative exists. Provide justification that other alternatives to cobalt based alloy have been evaluated and found unacceptable for AP600 applications.

252.32 Section 4.5.1 of the SSAR proposes to use Types 304 and 316 austenitic stainless steel in the control rod drive system. However, these materials are susceptible to intergranular stress corrosion cracking. Discuss why low carbon wrought austenitic stainless steel, which includes Types 304L, 316L, 304NG, 316NG, and modified 347, is not used instead.

252.33 Section 4.5.1 of the SSAR indicates that Inconel 750 and martensitic stainless steel Types 403 and 410 will be used in the control rod drive system. Verify that these materials are listed as acceptable in ASME Section III or Regulatory Guide 1.85, "Code Case Acceptability ASME Section III Materials." Otherwise, provide technical justifications for their use. In addition, discuss the heat treatment for these materials with technical justifications.

252.34 Discuss whether the carbon content of austenitic stainless steel will be limited to less than 0.02% in the control rod drive system as recommended in NUREG-0313, Revision 2, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," January 1988. Provide a technical discussion on why this limit is not relevant to the proposed use if it is not used (Section 4.5.1).

252.35 Discuss whether ferrite content limits for austenitic stainless steel castings and weld metal in the control rod drive system will be consistent with industry guidance (EPRI NP-6780-L) or staff guidance (NUREG-0313), whichever is more limiting. Provide technical justifications if these limits are not used (Section 4.5.1).

252.36 Section 4.5.1.1 of the SSAR indicates that materials in the control rod drive system are selected based on certain number of plant transients. For example, the SSAR assumes 320 reactor trips. However, the Standard Technical Specifications list 500 reactor trips for a plant with a 40-year design life. Demonstrate that the assumed plant transients in the SSAR are applicable to the projected 60-year plant design life (Section 4.5.1).

252.37 Provide information to confirm that the control rod drive materials are compatible with the reactor coolant as described in Articles NB-2160 and NB-3120 of the ASME Code (Section 4.5.1).

252.38 If precipitation-hardening stainless steels will be used in the control rod drive system, verify that these materials are listed as acceptable in ASME Section III or Regulatory Guide 1.85. Further, discuss the heat treatment for these materials with relevant technical bases (Section 4.5.1).

252.39 Discuss conformance of the control rod drive system with Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and Regulatory Guide 1.85. Provide technical justifications for any deviations, or provide acceptable alternatives (Section 4.5.1).

252.40 Section 4.5.1.3 of the SSAR indicates that Inconel 750 materials to be used in the control rod drive system will be ordered to specifications other than those in ASME Section III. Provide technical justifications that the alternative specifications meet the requirements in ASME Section III.

252.41 Section 4.5.1.4 of the SSAR indicates that the guidance in ASME NQA-2 will be used. However, ASME NQA-2 is not listed in Regulatory Guide 1.37 or 1.28, "Quality Assurance Program Requirements." Provide technical justifications for using ASME NQA-2.

Reactor Internals and Core Support Materials

252.42 Because Section 4.5.2 of the SSAR discusses both reactor internal and core support materials, consider revising the title of Section 4.5.2 of the SSAR accordingly.

252.43 Provide information to confirm that the materials selected for the construction of components of the reactor internals and core support structure will meet Section III of the ASME Code (Section 4.5.2).

252.44 Section 4.5.2.1 of the SSAR indicates that only a few materials will be used for the reactor internals and core supports. If other materials will also be used, identify them and address related concerns that have been raised on the control rod drive structural materials (see Q252.29 - Q252.41), if they are applicable to the material used for the reactor internals or core supports.

252.45 Section 4.5.2 of the SSAR proposes to use Types 304 and 316 austenitic stainless steel in the reactor internals and core support structures. However, these materials are susceptible to intergranular stress corrosion cracking. Justify why low carbon wrought austenitic stainless steel, which includes Types 304L, 316L, 304NG, 316NG, and modified 347, is not used instead.

252.46 Section 4.5.2 of the SSAR indicates that martensitic stainless steel Type 403 will be used in the reactor internals. Verify that this material is listed as acceptable in ASME Section III or Regulatory Guide 1.85. Otherwise, provide technical justifications for its use. In addition, discuss the heat treatment for this material with technical justifications.

252.47 Discuss whether the carbon content of austenitic stainless steel in the reactor internals and core support structures will be limited to less than 0.02% as recommended in NUREG-0313, Revision 2, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," January 1988. Provide a technical discussion on why this limit is not relevant to the proposed use if it is not used (Section 4.5.2).

252.48 Discuss whether ferrite content limits for austenitic stainless steel castings and weld metal in the reactor internals and core support structures will be consistent with industry guidance (EPRI NP-6780-L) or staff guidance (NUREG-0313), whichever is more limiting. Provide technical justifications if these limits are not used (Section 4.5.2).

252.49 Discuss conformance of the reactor internal and core support materials with Regulatory Guide 1.85 (Section 4.5.2).

252.50 Section 4.5.2.5 of the SSAR indicates that the guidance in ASME NQA-2 will be used with the reactor internals and core support structures. However, ASME NQA-2 is not listed in Regulatory Guide 1.37 or 1.28. Provide technical justifications for using ASME NQA-2.

Chapter 5

Reactor Coolant Pressure Boundary Materials

252.51 Table 5.2-1 of the SSAR lists "typical" material specifications for the reactor coolant pressure boundary (RCPB). Specify the actual materials for staff review.

252.52 Provide more details relating to the material specifications in Table 5.2-1 of the SSAR. For example, the reactor coolant piping is listed as SA376. However, SA376 can be further characterized by "type" with different properties. Section 3B.7.2 of the SSAR indicates that Type 316LN will be used (see Q252.15). Provide detailed information on RCPB materials in Table 5.2-1 of the SSAR.

252.53 Identify where Inconel 600 and 182 materials will be applied in the RCPB. Operating experience indicates that these materials are susceptible to cracking. If these materials will be used, provide technical information that demonstrates their suitability for the projected 60-year plant design life (Section 5.2.3).

252.54 Section 5.2.3.2.2 of the SSAR indicates that there may be carbon steel used in the RCPB. However, carbon steel materials may be susceptible to the mechanism of dynamic strain aging which reduces the material fracture properties (see Q252.1). Identify where carbon steel will be applied in the RCPB and discuss the procedures that will address the potential effects of dynamic strain aging.

252.55 Provide information to verify that the post-weld heat treatment discussed in the second paragraph in Section 5.2.3.2.2 of the SSAR meets the requirements in ASME Section III.

252.56 Recent fatigue test data indicate that the effects of the environment could significantly reduce the fatigue resistance of materials (see Q252.19). The specific concern relates to the reactor water and temperature environment and its synergistic interactions with strain rate. The recent data indicate that the design fatigue curves in ASME Section III Class 1 requirements may not be as conservative as originally intended. Describe the procedures that explicitly account for the effects of the environment in the fatigue analysis of components in the RCPB. (Cladding on base metal is not a structural material and should not be considered adequate to isolate the base metal from the effects of the environment. This is because the cladding may be breached, exposing the base metal to the water environment. Further, the cladding does not insulate the base metal from the reactor temperature.)

252.57 Section 5.2.3.3.1 of the SSAR indicates that the fracture toughness properties of the RCPB may meet the requirements of the ASME Code, Section III, Subsection NC. However, Subsection NC is for ASME Code Class 2 components, and is not applicable to the RCPB. Clarify your intent relating to the application of Subsection NC in the RCPB.

252.58 Section 5.2.3.3.1 of the SSAR indicates that Westinghouse has conducted a test program to show that the fracture toughness properties of low-alloy materials are "adequate." Demonstrate that the requirements of Subsection NB of Section III of the ASME Code are satisfied by the materials used in the RCPB.

252.59 Provide information in Section 5.2.3.3.1 of the SSAR to include an additional requirement that the fracture toughness of ferritic materials in the RCPB will meet Appendix G to 10 CFR Part 50.

252.60 Section 5.2.3.4.3 of the SSAR indicates that there may be inaccessible cavities or chambers in the RCPB. Discuss considerations to eliminate these conditions. If these conditions cannot be avoided, provide accesses for future inservice inspection to monitor the conditions in these cavities or chambers. Discuss the associated augmented inservice inspection program.

252.61 Section 5.2.3.4.3 of the SSAR excludes certain product forms from testing using ASTM A262. However, the test in ASTM A262 should be applicable to all product forms. Provide technical justifications for its proposed exclusions.

252.62 Section 5.2.3.4.5 of the SSAR indicates that there may be cast metals in the RCPB. However, cast stainless steel is subject to thermal aging (NUREG/CR-4513, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," June 1991) and is difficult to be inspected with ultrasonic techniques. Discuss considerations to use wrought materials instead, or demonstrate that the concerns over inspectability and thermal aging for the projected 60-year plant design life are alleviated.

252.63 Section 5.2.3.4.5 of the SSAR indicates that unstabilized austenitic stainless steel may be "retested". Provide additional information on the conditions for retesting.

252.64 Section 5.2.3.4.5 of the SSAR excludes certain materials from retesting. Provide technical justifications for these exclusions.

252.65 Section 5.2.3.4.6 of the SSAR indicates that the ferrite content for austenitic stainless steel weld metal will be controlled. Discuss whether ferrite content limits for austenitic stainless steel castings and weld metal will be consistent with industry guidance or staff guidance, whichever is more limiting (see Q 252.26). Provide technical justifications if these limits are not used.

252.66 Provide a discussion relating to lubricants for threaded fasteners within the RCPB. In particular, any application of molybdenum disulfide lubricants should be technically justified. Operating experience has indicated that molybdenum disulfide lubricants can cause stress corrosion cracking of fasteners (IE Bulletin 82-02, "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants," June 2, 1982) (Section 5.2.3).

252.67 Grinding of austenitic stainless steel materials may introduce susceptibility to stress corrosion cracking. EPRI Report NP-6780-L provides certain controls on grinding recommended by the industry. Describe the controls Westinghouse recommends be imposed on grinding (Section 5.2.3).

252.68 Discuss whether the carbon content of austenitic stainless steel in the RCPB will be limited to less than 0.02% as recommended in NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," Rev. 2, January 1988. Provide a technical discussion on the relevancy of this limit to the AP600 with the projected 60-year plant design life (Section 5.2.3).

252.69 Identify the application of electroslag welds in the RCPB (Section 5.2.3).

252.70 Confirm that the yield strength of cold-worked austenitic stainless steel in the RCPB will be less than 620 MPa (90 ksi) as recommended in Section 5.2.3 of the SRP (Section 5.2.3).

252.71 Discuss conformance of the AP600 with the guidance in NUREG-0313 as recommended in Acceptance Criterion II.2 of Section 5.2.3 of the SRP (Section 5.2.3).

252.72 Demonstrate that the nondestructive examination of ferritic steel and austenitic stainless steel tubular products in the RCPB will be in accordance with Section III of the ASME Code as recommended in Section 5.2.3 of the SRP (Section 5.2.3).

252.73 Section 1A of the SSAR discusses conformance of the AP600 design with regulatory guides. The SSAR proposes exceptions to Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants." The proposed alternative to Regulatory Guide 1.37 is based on staff guidance in Regulatory Guide 1.28, "Quality Assurance Program Requirements." Clarify the basis for the application of ASME NQA-2, which is not discussed in Regulatory Guide 1.28. Revise Section 1A of the SSAR accordingly, if appropriate.

252.74 Section 1A of the SSAR indicates conformance with the guidance in Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components." Discuss how Regulatory Position C.3 will be met. Specifically, clarify that a procedure qualification will be established in accordance with Regulatory Position C.2 even though Regulatory Position C.1 is not applicable. Describe whether Regulatory Position C.3 will be met if the production welding procedure does not conform to the qualified procedure. Revise Section 1A of the SSAR accordingly.

252.75 Section 1A of the SSAR proposes exceptions to Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." Provide technical justifications for not following the guidance in Regulatory Guide 1.44 or propose an acceptable alternative.

252.76 Section 1A of the SSAR proposes not to follow the guidance in Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility." Provide acceptable alternatives with the technical bases and revise Section 1A of the SSAR accordingly.

252.77 Because the activation of cobalt is a concern relating to the radioactivity in current nuclear plants, cobalt application should be avoided in AP600 for ALARA considerations. Identify the applications of cobalt based alloy in the RCPB. Demonstrate that other alternatives to cobalt based alloy have been evaluated and found unacceptable for AP600 applications (Section 5.2.3).

252.78 Discuss the limit on the cobalt content of all stainless steel and nickel based alloy RCPB components. Provide technical justifications if the limit exceeds 0.02 weight percent which is not consistent with industry guidelines (EPRI Report NP-6780-L) (Section 5.2.3).

252.79 Identify where Types 304 and 316 austenitic stainless steel will be applied in the RCPB. Because these materials are susceptible to intergranular stress corrosion cracking, discuss why low carbon wrought austenitic stainless steel, such as Types 304L, 316L, 304NG, 316NG, and modified 347, is not used instead (Section 5.2.3).

252.80 Section 5.3.2.7 of the SSAR indicates that the reactor vessel closure studs will be bricated from SA 540 materials. Identify the specific grade of the .erials. SA 540 Grade B23 or B24 materials have some of the high strengths among bolting materials permitted by Section III. High strength bolting materials may be susceptible to stress corrosion cracking. Provide technical justifications if the use of such high strength materials is being proposed.

252.81 Discuss the control on hardness of austenitic stainless steel during cold work fabrication operations, such as bending, cold forming, and straightening (Section 5.2.3).

Reactor Vessel Materials

252.82 Table 5.3-1 in the SSAR gives the same percentage of residual elements for the reactor vessel beltline forging and welds. Provide technical justifications for not lowering the residual element contents for the welds.

252.83 When the copper content of the reactor vessel beltline material is reduced, the susceptibility of the material to neutron irradiation may become dominated by other elements. Discuss the effects of not lowering the contents of nickel, phosphorous, and vanadium (Section 5.3).

252.84 Because the temperature affects the neutron embrittlement of the materials, provide information on the cold leg temperature. If a plant will operate at a cold leg temperature below 274°C (525°F), discuss the effects of temperature on embrittlement (Section 5.3).

252.85 Provide information to show that the reactor vessel materials will be heat-treated to achieve a fine grain microstructure (Section 5.3).

252.86 Table 5.3-3 in the SSAR shows the value for "RT_{PTS}" required by 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events." Provide details for this calculation, including assumptions and margins. The calculation should be based on the projected 60-year plant design life.

252.87 Westinghouse uses the guidance in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," to estimate the extent of neutron embrittlement. However, there are uncertainties in neutron embrittlement prediction procedures. For example, Regulatory Guide 1.99, Revision 1, would predict a reference temperature shift of 30°C (54°F) based on the phosphorous content, which is not addressed in Regulatory Guide 1.99, Revision 2. Thus, in calculating the shift in the reference temperature, Method 1 or Method 2 (as discussed below) should be used, whichever is more limiting:

Method 1:

A shift should be calculated based on Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials."

Method 2:

A shift should be calculated accounting for the phosphorous content and technical justifications for the methodology should be provided. Or, as an alternative, a shift may be estimated using the following equation:

$$A = [\ 40 + 1000\ (\ \%Cu - 0.08\ ) + 5000\ (\ \%P - 0.008\ )\ ]\ [\ f / 10^{19}\ ]^{1/2}$$

where

A = predicted shift, °F

f = fluence, n/cm^2 (E>1 MeV)

$\%Cu$ = weight percent of copper (If $\%Cu \leq 0.08$, use 0.08.)

$\%P$ = weight percent of phosphorus (If $\%P \leq 0.008$, use 0.008.)

Describe how this approach in estimating the reference temperature shift is met (Section 5.3).

252.88 Table 5.3-3 in the SSAR lists gross bounds on the effects of neutron embrittlement on the reactor vessel materials. Provide details of the results, not gross bounds, and the calculation procedures, such as assumptions and margins used. Show the "RT_{NDT}" for the inner surface of the vessel and the "RT_{NDT}" and the upper shelf energy (transverse direction) for both the inner surface and "1/4-T" location as discussed in Section 5.3.2.4 of the SSAR. The result should be based on the projected 60-year plant design life. Further, the shift in the reference temperature should be calculated as described in Q252.80.

252.89 Section 5.3.2.4 of the SSAR discusses conformance with regulatory guides. The applicant should also discuss conformance with Regulatory Guide 1.37.

252.90 Section 5.3.2.5 of the SSAR indicates that the minimum initial upper shelf energy of the reactor vessel beltline materials will be 102 J

(75 ft-lb). Provide information to confirm that the applicable upper shelf energy is that measured for the transverse direction of the materials.

252.91 Section 5.3.5 of the SSAR indicates that the reactor vessel materials surveillance program will be in accordance with ASTM E185-83. However, the applicable version of ASTM E185 that is referenced in Appendix H to 10 CFR Part 50 is ASTM E185-82. Demonstrate how Appendix H to 10 CFR Part 50 is met.

252.92 The reactor vessel materials surveillance program depends on the estimated shift of the reference temperature according to ASTM E185-82. For establishing the surveillance program, estimate the shift in the reference temperature using the following methods:

(i) Method 1 discussed in Q252.80.

(ii) Method 2 discussed in Q252.80.

(iii) A shift should be assumed to be greater than 56°C (100°F) but less than 111°C (200°F).

The shift estimate should be based on Item (i), (ii), or (iii), whichever results in the largest temperature shift.

Because of uncertainties in current methods in estimating neutron embrittlement, the staff has established a minimum shift estimate in Item (iii) in developing a surveillance program for design certification. The staff concludes that the reactor vessel materials surveillance program plan should be based on a reasonably conservative estimate of the temperature shift. This is because it may be technically difficult to backfit an existing surveillance program should the actual temperature shift be higher than that estimated.

Describe how this approach in estimating the shift in the reference temperature for the surveillance program is met (Section 5.3).

252.93 Appendix H to 10 CFR Part 50 requires the reactor vessel materials surveillance program to meet ASTM E185-82. ASTM E185-82 has been applicable to plants designed for 40 years, i.e., 32 effective full-power years (EFPYs) at end-of-life. Thus, the staff finds that the schedule in ASTM E185-82 should be maintained for 40 years (32 EFPYs).

Further, the schedule in ASTM E185-82 should be supplemented to address the period between 40 and the projected 60 years for the AP600. Propose a capsule withdrawal schedule beyond 32 EFPYs to demonstrate compliance with Appendix H of 10 CFR Part 50 to the end of the AP600's proposed design life of 60 years. One option may be to maintain the time interval between the last two capsule withdrawals within 32 EFPYs throughout the rest of plant design life, or at the end of the proposed 60-year plant design life, whichever is earlier.

For example, if a design certification applicant estimates that the reference temperature shift is greater than 56°C (100°F) but less than 111°C (200°F) using the procedures discussed in Q252.85, the capsule withdrawal schedule in ASTM E185-82 would require one capsule each to be withdrawn at 3, 6, 15, and 32 EFPYs along with certain restrictions on fluence levels. This schedule should be followed for up to 32 EFPYs. In addition, propose a schedule beyond 32 EFPYs to the end of the proposed 60-year plant design life (Section 5.3).

252.94 Provide information on the inclusion of standard reference materials in its surveillance capsules (Section 5.3).

252.95 Describe whether weld metals and weld heat-affected zone (HAZ) materials will be included in the surveillance program. If not, provide technical justifications for the non-inclusion (Section 5.3).

252.96 Describe the "lead factors" for the surveillance capsules (Section 5.3).

252.97 Discuss design provisions for the installation of replacement surveillance capsules (Section 5.3).

252.98 Section 5.3.4.6 of the SSAR indicates that there is a Table 5.3-7 in the SSAR. The staff cannot find this table. Correct or clarify this reference.

252.99 Section 5.3.4.7 of the SSAR discusses the acceptance criterion for cladding bond defects during reactor vessel fabrication. Provide technical justifications for the acceptance criterion.

252.100 Describe the lubricant to be used on the reactor vessel closure head studs and provide technical justifications. The staff's concern relating to the application of lubricants containing molybdenum disulfide has been discussed in Q252.59 (Section 5.3).

252.101 The staff's concern relating to the environmental effects on fatigue has been discussed in Q252.19 and Q252.49. This is applicable to all materials. Address this concern for the reactor vessel materials (Section 5.3).

252.102 Discuss design considerations for facilitating an in-place reactor vessel thermal annealing treatment should this become necessary (Section 5.3).

Pressure-Temperature Limits

252.103 Figures 5.3-2 and 5.3-3 in the SSAR show the heatup and cooldown pressure-temperature curves for the reactor vessel. Discuss whether these curves will be the actual curves for the plant.

252.104 There are uncertainties in neutron embrittlement prediction procedures. Thus, for establishing the reactor vessel pressure-temperature limits prior to the availability of valid plant specific surveillance data, estimate the shift in the reference temperature using either Method 1 or 2 as described in Q252.80, whichever is more limiting. The reference temperature shift should be based on the proposed design life of 60 years. The COL applicant will be requested to commit to reviewing the continued applicability of the pressure-temperature limits when plant specific surveillance data become available. Provide information to show that this approach in establishing pressure-temperature limits is met (Section 5.3.3).

252.105 Provide details for the pressure-temperature limit calculations, including assumptions and margins. Estimate the shift in the reference temperature according to Q252.97. Further, identify any deviations from the recommended calculational procedures in Section 5.3.2 of the SRP (Section 5.3.3).

252.106 Demonstrate that its pressure-temperature limits are in accordance with Appendix G to 10 CFR Part 50. For example, verify that the limit for the closure flange is satisfied (Section 5.3.3).

Steam Generator Materials

252.107 The proposed new steam generator tube plugging criteria in Section 5.4.2 of the SSAR would place increased emphasis for steam generator integrity on primary to secondary leakage monitoring relying on increased sensitivity and on-line real time read-outs. Describe Westinghouse's proposed plans on implementing this monitoring.

252.108 Describe how the "Delta-75" steam generator design proposed for the AP600 will facilitate the implementation of in-situ fusion techniques for steam generator tube repair. Also, discuss how the selection of materials for the tube support structures and the tubesheet will preclude deleterious effects on material toughness caused by in-situ fusion heat effects (Section 5.4.2).

252.109 Section 5.4.2.3.3 of the SSAR indicates that tube vibration has potential to cause wear. Discuss in detail the potential for wear degradation with emphasis on the AP600 features that are designed to mitigate this concern.

252.110 Section 5.4.2.3.3 of the SSAR discusses flow-induced vibrations with special emphasis on fluid elastic vibration. Provide the results of prototype tests and calculations to support the discussion.

252.111 Recent plant operating experience disclosed the possibility of miss-placed anti-vibration bars (AVBs) and the possible severe consequences. Discuss how the proper location of AVBs will be ensured (Section 5.4.2).

252.112 Industry recommendations and other vendors' improved steam generators designs incorporate primary side manways having a minimum inner diameter of 53 cm (21 in). Discuss Westinghouse's technical basis for limiting the ports in the "Delta-75" steam generator to 46 cm (18 in) in diameter as indicated in Section 5.4.2.5 of the SSAR.

252.113 Experience has shown the advisability of complete records and archive materials to investigate corrosion and mechanical damage which may occur during service. Industry recommendations suggest archiving at least 2 m (7 ft) of each heat of row 1 and 2 "U-bends" prior to final heat treatment and following the mill anneal, and production samples containing tubes from each heat expanded in a tube sheet mockup. Archive samples should be maintained to support future chemical cleaning programs and for possible defect calibration samples for inservice inspection. Describe Westinghouse's program to retain records and archive materials (Section 5.4.2).

252.114 Provide detailed discussion on the extensive operating experience and laboratory testing (including model boiler tests) to justify the use of all volatile treatment (AVT) secondary water chemistry with Inconel 690 for the proposed 60-year plant design life (Section 5.4.2).

252.115 Address the potential for primary water stress corrosion cracking in Inconel 690 for the proposed 60-year plant design life (Section 5.4.2).

252.116 Address the resistance to corrosion of Inconel 690 in upset water chemistry conditions which would take place over the proposed 60-year plant design life (Section 5.4.2).

Chapter 6

Engineered Safety Features Materials

252.117 Confirm that the 1989 edition, 1989 addenda, of the ASME Section III Code is the applicable code for the materials used in the engineered safety features of the AP600 (Section 6.1.1).

252.118 Section 6.1.1.1 of the SSAR discusses "principal" materials for the engineered safety features (ESF). Provide information on all materials in the ESF.

252.119 Table 6.1-1 in the SSAR lists materials for the ESF. However, this list lacks specificity, e.g., it lists "austenitic stainless steel." This list also refers to other sections of the SSAR where information may not be readily available. For example, it lists the passive containment cooling system water storage tank in Section 3.8.4 of the SSAR. But the information on the materials cannot be found there. Further, this list may not be complete. Revise Table 6.1-1 in the SSAR to provide more specific information regarding materials used in the ESFs.

252.120 Section 1A of the SSAR indicates that the guidance in Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," will not be applied to ASME Code Class 2 and 3 components in the AP600. Provide technical justifications for not following this guidance or provide acceptable alternatives to this guidance. In addition, provide confirmation that preheat requirements in the ASME Section III Code will be satisfied. Revise Section 1A of the SSAR accordingly.

252.121 Describe the moisture control on low hydrogen welding materials (Section 6.1.1).

252.122 Indicate conformance with Regulatory Guides 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," and 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," as related to Section 6.1.1 of the SRP. Provide technical justifications if the guidance contained in these documents will not be followed, or provide alternatives with bases to demonstrate the equivalency (Section 6.1.1).

252.123 Section 6.1.1.1 of the SSAR indicates that certain materials produced under ASTM designations are acceptable. Provide technical justifications that these materials satisfy the ASME Section III Code requirements.

252.124 Discuss welding requirements for areas of limited accessibility (Section 6.1.1).

252.125 Provide a corrosion allowance for the materials used in the engineered safety features of the AP600 for the projected 60-year plant design life along with the technical basis to support the allowance (Section 6.1.1).

252.126 Discuss hydrogen generation from the corrosion of materials within the containment, such as aluminum and zinc, based on an assumed, justified corrosion rate (Section 6.1.1).

252.127 Section 6.1.1.6 of the SSAR indicates that the AP600 design conforms to Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants." However, Section 1A of the SSAR indicates that Westinghouse is proposing exceptions to Regulatory Guide 1.3 Provide information to clarify your intent. Revise Section 1A of the SSAR accordingly.

252.128 Discuss whether the carbon content of austenitic stainless steel used in the ESFs will be limited to less than 0.02% as recommended in NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," Rev. 2, January 1988 (Section 6.1.1).

252.129 Discuss whether ferrite content limits for austenitic stainless steel castings and weld metal used in ESFs will be consistent with industry guidance (EPRI Report NP-6780-L) or staff guidance (NUREG-0313), whichever is more limiting. Provide technical bases if these limits are not used (Section 6.1.1).

252.130 Carbon steel materials may be susceptible to the mechanism of dynamic strain aging which reduces the material fracture properties (see Q252.18). Describe procedures to address the effects of dynamic strain aging for materials used in the ESFs (Section 6.1.1).

252.131 Identify where Types 304 and 316 austenitic stainless steel are applied in the ESFs. Because these materials are susceptible to intergranular stress corrosion cracking, discuss why low carbon wrought austenitic stainless steel, which includes Types 304L, 316L, 304NG, 316NG, and modified 347, is not used instead (Section 6.1.1).

252.132 Identify where Inconel 600 and 182 are applied in the ESFs. Operating experience indicates that these materials are susceptible to cracking. If these materials will be used, discuss any special measures to be taken to reduce the susceptibility to cracking and provide test data to demonstrate that the materials are not susceptible to cracking for the projected 60-year plant design life (Section 6.1.1).

252.133 Recent fatigue test data indicate that the effects of the environment could significantly reduce the fatigue resistance of materials (see Q252.19) The specific concerns relate to the reactor water and temperature environment and its synergistic interactions with the strain rate. Describe the procedures that explicitly account for the effects of the environment in the fatigue analysis of materials used in the ESFs (Section 6.1.1).

Chapter 9

Demineralized Water Makeup System

252.134 Describe the materials of construction for the major components of the demineralized water treatment system such as pumps, valves, and piping (Section 9.2.3).

252.135 Although the demineralized water system does not perform any safety related function, describe whether the design of the system ensures that failure of any of its component would not jeopardize performance of the systems required for safe plant shutdown (Section 9.2.3).

Steam and Feedwater System Materials

252.136 Confirm that the 1989 edition, 1989 addenda, of the ASME Section III Code is the applicable code for the materials used in the steam and feedwater system of the AP600 (Section 10.3.6).

252.137 Identify the steam and feedwater system materials and provide information to demonstrate that the materials meet the requirements of Section III of the ASME Code (Section 10.3.6).

252.138 Provide information to indicate that the tubular products in the steam and feedwater system will be examined in accordance with Section III of the ASME Code (Section 10.3.6).

252.139 Provide a corrosion allowance for the steam and feedwater system for the projected 60-year plant design life along with the technical basis for the allowance (Section 10.3.6).

252.140 Discuss provisions to address the potential for erosion/corrosion of the steam and feedwater system. Justify that erosion/corrosion will be insignificant for the projected 60-year plant design life (Section 10.3.6).

252.141 Carbon steel materials may be susceptible to the mechanism of dynamic strain aging which reduces the material fracture properties (see Q252.18). Describe procedures to address the effects of dynamic strain aging for materials used in the steam and feedwater system (Section 10.3.6).

252.142 Recent fatigue test data indicate that the effects of the environment could significantly reduce the fatigue resistance of materials (see Q252.19). The specific concerns relate to the reactor water and temperature environment and its synergistic interactions with the strain rate. Describe the procedures that explicitly account for the effects of the environment in the fatigue analysis of materials used in the steam and feedwater system (Section 10.3.6).

Condensate Cleanup System

252.143 Although the condensate polishing system serves no safety related function, show that failure of any of its components will not cause damage to the systems required for safe plant shutdown (Section 10.4.6.1.1).

252.144 Describe safety provisions that will be taken in the event of radioactive contamination of the fluids handled by the condensate polishing system in order to meet the ALARA requirements (Section 10.4.6).

Steam Generator Blowdown System

252.145 Describe the materials of construction of different components in the steam generator blowdown system (Section 10.4.8).

CHEMICAL ENGINEERING

Chapter 5

281.1 Table 5.2-2 of the SSAR lists "recommended" reactor coolant system (RCS) water chemistry specifications. Specify the actual RCS water chemistry.

281.2 Section 5.2.3.2.1 of the SSAR discusses the primary water chemistry for AP600. Is the RCS water chemistry consistent with the guidelines of EPRI Reports NP-6780-L and NP-7077, "PWR Primary Water Chemistry Guidelines: Revision 2," November 1990 that are identified in Chapter 1 of the ALWR Utility Requirements Document for passive plants, Volume III? Identify differences between the primary water chemistry of the AP600 and these guidelines, and provide justification for the deviations.

Chapter 6

281.3 Demonstrate that the composition of containment spray and core cooling water will be controlled to ensure a minimum pH of 7 (Section 6.1.1).

281.4 Discuss coatings in Section 6.1.1 of the SSAR in accordance with Acceptance Criterion II.B.4 of Section 6.1.1 of the SRP.

281.5 Provide information in Section 6.1 of the SSAR to demonstrate that the pH for the emergency coolant water will comply with the Branch Technical Position MTEB 6-1. Otherwise, provide technical justifications for deviations from this position.

281.6 Discuss compliance of protective coatings (organic materials) with the quality assurance requirements of Appendix B to 10 CFR Part 50 (Section 6.1.2).

281.7 Discuss conformance protective coatings (organic materials) with the guidance in Regulatory Guide 1.54 and provide technical justifications for any deviations (Section 6.1.2).

281.8 Provide technical justifications for not using ANSI Standard N101.2 or propose an acceptable alternative (Section 6.1.2).

281.9 Provide information to justify its assumption that the elemental and particulate iodine released during an accident could be satisfactorily removed from the containment atmosphere by surface deposition and sedimentation without use of the containment spray (Section 6.5.2).

Chapter 9

Demineralized Water Makeup System

281.10 Describe why the guidelines in Section 9.2.3.1.2 of the SSAR for demineralized water do not include specifications for halogens and sulfate.

Process and Post-Accident Sampling Systems

281.11 Section 9.3.3.1.2.2 of the SSAR contains a statement that the design of the post-accident sampling system (PASS) complies with NUREG-0737 and NUREG-4330. Since these two documents contain different recommendations, describe which of the two documents will be used as the basis for design and operation of the PASS in the AP600 plant (Section 9.3.2).

281.12 NUREG-0737 requires the PASS to have capability for sampling the containment atmosphere for the radionuclides that may be indicators of the degree of core damage, e.g., noble gases. Sections 9.3.3.2.2 and 9.3.3.4.2 of the SSAR indicate departure from this requirement in the AP600 design. Provide technical justifications for the deviations.

281.13 Describe how the proper operation of the PASS will be verified. In addition, discuss the inspection and testing requirements (Section 9.3.3.6).

Chemical and Volume Control System

281.14 Describe the maximum steam generator tube leak that can be accommodated by the chemical and volume control system (CVCS) makeup pumps (Section 9.3.6.1.2.2).

281.15 Identify the location where the hydrotest pump will be attached to the CVCS and discuss provisions to ensure that the system will withstand the pressure generated by this pump (Section 9.3.6.1.2.5).

281.16 Provide a description of the mixed bed and cation bed demineralizers. In addition, discuss provisions for spent resin regeneration (Section 9.3.6.2.1.1).

281.17 Describe the safety precautions for storing the hydrogen used for oxygen control in the reactor coolant (Section 9.3.6.2.4).

281.18 Does the safety analysis of the plant takes credit for the injection flow produced by the CVCS makeup pumps during an accident? If credit is taken for this injection, the CVCS should be considered a safety-related system and this would contradict the definition of the system made in Section 9.3.6.1.1 of the SSAR (Section 9.3.6.2.4).

281.19 Explain why the hydrogen supply line, which is directly connected to the reactor coolant water purification loop in the CVCS and penetrates the reactor containment boundary, has only one isolation valve. It should have two isolation valves such as in the letdown and makeup lines in the CVCS, as it is required by General Design Criterion 55 (Section 9.3.6.3.7).

RADIATION PROTECTION

471.1 Provide information on dimensions, volumes, and wall material compositions for radiation source containing components in the plant. Provide information on the material composition and thicknesses of shield walls around these radioactive sources. This information is necessary for the staff to perform confirmatory shielding calculations to determine dose rates in potentially occupied areas adjacent to these components.

471.2 Section 12.3 of the SRP specifies that the SSAR should contain radiation zone designations (including zone boundaries and normal traffic patterns) on the plant layout drawings. The zone maps are laid out very well, except that Very High Radiation Areas as defined in the revised 10 CFR Part 20 are not identified during normal and anticipated operational occurrences. Also, there are no traffic patterns identified for normal traffic flow or for access to vital areas during accident operations. This information is needed by the staff to ensure all areas having potentially lethal levels of radiation are identified and controlled. Provide this information.

471.3 Provide expected peak airborne radioactivity concentrations, estimated man-hours of occupancy, and estimated inhalation exposures for all areas of the plant accessed by plant personnel. This information is required by the Standard Review Plan and is needed by the staff to ensure that the plant's ventilation flow is sufficient to maintain airborne radioactivity levels ALARA.