UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 29, 1994

Docket No. 52-003

Mr. Nicholas 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 area of mechanical engineering (Q210.29-Q210.110).* The staff positions identified in the enclosure represent positions identified in published review guidance (such as the standard review plan or regulatory guides), positions taken during the staff's review of the evolutionary design certification applications or the EPRI ALWR Requirements Document, or positions developed during the staff's review of the unique features of the AP600 design. Justification for any proposed deviation from these positions should be provided with your responses.

Enclosed are the staff's questions. Please respond to this request by June 30, 1994 to support the staff's review of the AP600 design.

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.

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

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This request for additional information affects nine or fewer respondents, and therefore is not subject to review by the Office of Management and Budget under P.L. 96-511.

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

Sincerely,

(Original signed by)

Thomas J. Kenyon, Project Manager
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Enclosure:
As stated

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REQUEST FOR ADDITIONAL INFORMATION ON THE WESTINGHOUSE AP600 DESIGN

210.29 With respect to quality group classification of certain systems, components, and equipment, the staff does not completely agree with the information in Section 3.2.2.5 of the SSAR, "Equipment Class C;" Appendix 1A of the SSAR, "Conformance with Regulatory Guides;" the response to Q 210.1 (dated December 22, 1992); and the exceptions to Section 3.2.2 of the Standard Review Plan (SRP) in Revision 1 of WCAP-13054.

Section 3.2.2.5 of the SSAR states that items that perform one or more of the following safety-related functions are classified as Class C (Quality Group (QG) C):

- Provide safety-injection or maintain sufficient reactor coolant inventory to allow for core cooling
- Provide core cooling
- Provide containment cooling
- Provide for removal of radiation from the containment atmosphere as necessary to meet the offsite dose limits

To be consistent with Regulatory Guide (RG) 1.26 and ANSI/ANS 51.1, the staff's position is that all items that perform one or more of the above functions should be Class B (QG B).

In Appendix 1A of the SSAR and WCAP-13054, exception is taken to Position C.1.a of RG 1.26, which is the basis for the staff position stated above relative to Section 3.2.2.5. The basis stated in the SSAR for this exception is that for the AP600, QG B is reserved for the containment boundary including the containment isolation valves. The exception acknowledges that for QG C, the ASME examination and inservice inspection rules are less stringent than those for QG B. The SSAR states that QG C is acceptable for passive safety-systems components such as the accumulators and the IRWST, and that minor leakage is not a problem because (a) these components are inside containment, (b) minor leakage does not affect the component's functional performance, and (c) there is continuous water level and gas pressure monitoring of the accumulators that detects leaks. The following is the staff's position relative to this exception:

As stated in Section 6.3 of the SSAR, the primary function of the passive core cooling system is to provide emergency core cooling following postulated design basis events. RG 1.26 classifies emergency core cooling systems as QG B. The system boundary includes those portions of the system required to accomplish the specified safety function, and connected piping up to and including the first valve that is either closed or capable of automatic closure when the safety function is required. This is irrespective of the fact that the system does not recirculate post-accident fluid outside containment. In addition, the position stated in the SSAR for the exception relative to the ability of the accumulators and the IWRST to accommodate minor leakage appears to be similar to the piping leak-before-break (LBB) issue, but without a technical basis to implement LBB for these components. The staff cannot accept such an argument as the basis for the design of an emergency core cooling system. Therefore, the staff's position is that these components should be QG B.

The response to Q210.1 contains positions similar to those in Section 3.2.2.5 and Appendix 1A of the SSAR, and is not acceptable for the same reasons discussed above. Therefore, revise the applicable information in Appendix 1A relative to RG 1.29, Section 3.2.2.5, Table 3.2-3, Section 6.3 (including the P&IDs), WCAP-13054, and the response to Q210.1 to comply with the above staff positions.

210.30 The response to Q210.8 dated December 22, 1992 is not completely acceptable. At the request of the Nuclear Management and Resources Council (NUMARC), the staff's review of EPRI NP-6628 has been put on hold pending a decision by NUMARC relative to the continuation of this review. To date, the staff has not accepted an experience-based approach for the seismic design of safety-related piping systems. Therefore, the staff's position remains that EPRI NP-6628 is currently not acceptable. Revise Section 3.7.3.8.2.2 of the SSAR and the response to Q210.8 to remove the reference to this report.

210.31 The response to Q210.9 dated January 22, 1993 relative to the jurisdictional boundary between module frameworks and piping supports within the module appears to be acceptable. However, the staff does not agree with the last paragraph in this response, which states that subsequent to the incorporation of AISC N690 into the ASME Code, the design criteria for linear supports would change from Subsection NF to AISC N690. Subsection NF includes rules for construction of such supports, where "construction" is as defined in Subsection NF-1100(a).

Therefore, even after an acceptable incorporation of AISC N690 into the Code, the staff's position is that Subsection NF, not AISC N690, will remain the only staff-endorsed rules for these supports. Revise the response to Q210.9 to delete the last paragraph.

210.32 The response to Q210.12 dated November 30, 1992 references EPRI NP-6153, "Seismic Analysis of Multiply Supported Piping Systems," as the basis for combining the results of the modal spectra analysis and seismic anchor motion by the square root sum of the squares (SRSS) method. The staff has not endorsed EPRI NP-6153 and does not agree that this report provides an adequate technical basis for using the SRSS method. The staff's position remains as stated in Q210.12, i.e., the responses due to the inertia effect and SAM should be combined by the absolute sum method (see Section 3.9.2.II.2.g of the SRP). Revise Section 3.7.3.9 of the SSAR to reflect this staff position. In addition, either revise of delete the exception to Section 3.9.2.II.2.g of the SRP in WCAP-13054.

210.33 The response to Q210.15 dated December 22, 1992 requires more detailed information relative to the methods for verification of computer programs. Section 3.9.1.2 of the SSAR and the response to Q210.15 both reference Chapter 17 of the SSAR, "Quality Assurance," for this information. However, Chapter 17 does not contain the level of detail that the staff is seeking. As a minimum, the staff requests that each program used in dynamic and static analyses to determine the structural and functional integrity of seismic Category I ASME Code and non-code items for the AP600 be verified by at least one of the following methods:

1. Hand calculations
2. Analytical results published in the literature
3. Acceptable experimental tests
4. Results from a similar program previously endorsed by the staff
5. Comparison with the benchmark problems in NUREG/CR 1677, "Piping Benchmark Problems."

The following programs listed in Table 3.9.15 of the SSAR have been reviewed and endorsed by the staff:

ABAQUS	Finite element structural analysis
ANSYS	Finite element structural analysis
GAPPIPE	Static and dynamic analysis of piping systems
WECAN	Finite element structural analysis
Westdyn	Static and dynamic analysis of piping systems

For the remainder of the programs in Table 3.9.15 and all other applicable programs that will be listed in the ASME Code Design Reports, revise Section 3.9.1.2 of the SSAR to identify one or more of the above verification methods. In addition, delete the exception to SRP 3.9.1, Section II.2, in Revision 1 to WCAP-13054.

210.34 Table 3.2-3 of the SSAR, "AP600 Classification of Systems and Components" does not appear to include the classifications for the piping and supports of each system. Although these classifications may be implicit in this table, the staff's position is that this important table should explicitly include the classifications, principle construction codes, and quality assurance programs for all piping and all supports for piping and equipment in each system. The supports should have the same classification as the supported component or equipment. This information should be consistent with applicable piping and instrumentation diagrams in the SSAR. Revise Table 3.2-3 to include this information.

210.35 Quality assurance programs should be identified in Table 3.2-3 of the SSAR. Revise Table 3.2-3 to add a column that identifies the quality assurance requirements for each line item in the table.

210.36 In Table 3.2-3, Sheets 5 through 8, and Figure 9.2.2-2, Sheet 2, of the SSAR, the equipment and piping in the Component Cooling Water System (CCS) is all classified as QG D, with the exception of the containment penetration area. Section 9.2.2 of the SSAR states that the CCS serves no safety-related function except for containment isolation. However, Table 9.2.2-2 of the SSAR lists the following safety-related components for which the CCS provides a reliable supply of cooling water:

- Reactor coolant pumps (ASME Class 1)
- Chemical Volume and Control System letdown heat exchangers (ASME Class 3)
- Normal RHR heat exchangers and pumps (ASME Class 3)

Position 2b of RG 1.26 states that portions of cooling water systems that perform functions similar to those above should be classified as QG C. This means that the design of these components should meet all of the rules of ASME Class 3. Either revise Table 3.2-3, Section 9.2.2, Figure 9.2.2-2, and applicable portions of Appendix 1A of the SSAR and Revision 1 to WCAP 13054 to reflect this staff position, or provide a detailed justification for the AP600 position on this issue.

210.37 Table 3.2-3 and Figure 5.4-7 of the SSAR, "P&ID for the Normal Residual Heat Removal System (RNS)," identifies all components within the RNS as seismic Category I and either AP600 Class A, B, or C (ASME Class 1, 2, or 3), with the principal construction code as ASME NB, NC, or ND, respectively. The staff's interpretation of these commitments is that RNS components will be constructed in accordance with ASME Subsections NB, NC, and ND, as applicable, where construction is as defined in NB-1110(a) or NC/ND-1100(a). In addition, applicable ASME Section XI rules will apply. However, Section 5.4.7.1.2 of the SSAR states that the RNS piping and components are Safety Class C, seismic Category I for pressure retention purposes only. This statement does not appear to be consistent with the staff's interpretation of Table 3.2-3 and the P&ID. Revise Section 5.4.7.1.2 to be consistent with Table 3.2-3 and Figure 5.4-7.

210.38 Section 5.4.14 of the SSAR states that the passive residual heat removal heat exchanger (PRHR HX) is AP600 Equipment Class A and its supports are Class C. These supports are not listed in Table 3.2-3 and are not included in Figure 6.3-2 of the SSAR, "P&ID for Passive Core Cooling System." Figure 5.4-9 provides a sketch of these supports, but there is not enough detail to understand the support configuration. The staff's position is that supports for safety-related components and equipment should be the same safety class as the supported item. Revise the section, table, and figures to classify the supports for the PRHR HX as Equipment Class A, seismic Category I.

210.39 Section 6.3.2.2.5 of the SSAR states that the PRHR HX inlet and outlet piping connects to inlet and outlet channel heads mounted through the In-containment Refueling Water Storage Tank (IRWST) wall with a tubesheet that is part of the tank wall. The PRHR HX is Class A, and the IRWST is Class C. Section 3.8.3.1.7 states that the east wall of the IRWST consists of structural steel modules, filled with concrete and forming, in part, the refueling cavity, steam generator compartment, and pressurizer compartment walls. It is not clear to the staff what the relationship is between the IRWST tank wall, tubesheets, channel heads, and the structural steel modules. Appendix 3A and Figure 6.3-2 of the SSAR do not appear to contain this information. Revise the SSAR to provide a complete description of this area. Include sufficient sketches and information to (a) describe the above relationships, (b) identify the AP600 Equipment Safety Class of each item and clearly identify the location of the interface between the Class A PRHR HX and the Class C IRWST, and (c) describe the details of heat exchanger inlet and outlet piping pass-throughs in the modules, including a description of applicable analyses.

210.40 Section 3.6.2.1.1.4 of the SSAR, "High Energy Piping in Containment Penetration Areas" should be changed as follows to be consistent with staff positions in Section 3.6.2 of the SRP:

a. Revise the third bullet to include a commitment that when guard pipes are used in this area, the enclosed portion of fluid system piping should not only be seamless, but should not contain circumferential welds unless specific access provisions are made in the guard pipe to permit inservice volumetric examination of these welds in accordance with the augmented inservice inspection provisions in the fourth bullet of this subsection. If applicable, inspection ports in the guard pipe should not be located in that portion of the guard pipe passing through a shield building annulus.

b. The definition of break exclusion areas in the first paragraph and last three bullets of this subsection are not completely acceptable. The staff's position on this issue is that this area can start at the inboard side of the inside isolation valve but must end at the outboard side of the outside isolation valve. Revise these portions of this subsection to be consistent with the staff position.

210.41 Section 3.6.2.3.1 of the SSAR, "Jet Impingement," states that if a simplified static analysis is performed instead of a dynamic analysis, the jet impingement force is multiplied by a dynamic load factor of 1.2 to 2.0, depending upon the time variance of the jet load. The staff's position, which agrees with Section 7.3 of ANSI/ANS 58.2-1988 (Ref. 4 in Section 3.6.4 of the SSAR) is that this load factor should be 2.0. Either revise Section 3.6.2.3.1 to reflect this staff position, or provide a more detailed basis for a 1.2 factor.

210.42 Sections 3.6.2.3.2, 3.9.3.4, and 3.10.1.3 of the SSAR mention an analysis approach for transient loading conditions and Service Level D conditions that allows a limited number of pipe supports to fail, provided that the consequences of these failures are evaluated and that adequate support exists for deadweight and steady state pressure conditions following the event. Since these are ASME Class 1, 2, or 3 supports, they are designed to ASME Subsection NF, and the loading combinations in Table 3.9-8 of the SSAR and, therefore, should withstand Service Level D loads without failure. What is the advantage of postulating failures of such supports? Provide a more detailed discussion of this procedure and how it will be implemented.

210.43 Section 3.6.2.3.4.2 of the SSAR states that if energy absorbing material is used in the design of pipe whip restraints, the allowable deflection is 80% of the maximum crushable height at uniform crushable strength. In accordance with Section 3.6.2.III.2.a of the SRP, the staff's position is that the allowable capacity of crushable material shall be limited to 80% of its rated energy dissipating capacity as determined by dynamic testing at loaded rates within ±50% of the specified design loading rate. The rated energy dissipating capacity shall be taken as not greater than the area under the load-deflection curve as illustrated in Figure 3.6.2-1 of Section 3.6.2 of the SRP. Revise Section 3.6.2.3.4.2 to be consistent with the staff's position.

210.44 Section 3.6.2.4 of the SSAR, "Protective Assembly Design Criteria," states that auxiliary guard pipes provide additional confidence that pipes will not leak into the annulus between the containment wall and the shield building. This implies that these guard pipes are identical to those in the containment penetration area break exclusion zone which are discussed in Section 3.6.2.1.1.4. However, Section 3.6.2.4.2 of the SSAR, "Auxiliary Guard Pipes," provides design criteria for these guard pipes that is not consistent with the criteria in Section 3.6.2.1.1.4, and is unacceptable for guard pipes in the break exclusion zones. Revise Sections 3.6.2.4 and 3.6.2.4.2 to more clearly define the difference, if any, between auxiliary guard pipes and those in the break exclusion zones, and identify more specifically where auxiliary guard pipes will be used in the AP600.

210.45 Section 3.6.2.4.2 of the SSAR states that auxiliary guard pipes will be constructed in accordance with the rules for ASME Section III, Class 3 piping. Because of potential in-service inspection problems, the staff discourages the use of auxiliary guard pipes. However, if the response to Q210.44 indicates that they will be used, the staff's position is that they should be constructed to the same ASME rules as those required for the enclosed piping. Revise Section 3.6.2.4.2 to reflect this staff position.

210.46 Section 3.7.3.1 of the SSAR states that one of the methods used for seismic analysis is "design by rule." Revise this section to define this term and to reference those sections of the SSAR which contain design by rule methods.

210.47 Revise Section 3.7.3.4 of the SSAR, "Basis for Selection of Frequencies," to include a commitment to the guidelines of Section 3.9.2.II.2.c of the SRP, i.e., to avoid resonance, the fundamental frequencies of components and equipment should preferably be selected to be less that 1/2 or more than twice the dominant frequencies of the support structure.

210.48 The following requests are relative to Section 3.7.3.5 of the SSAR, "Equivalent Static Load Method of Analysis:"

a. The second paragraph in this section states that single degree of freedom subsystems are designed for accelerations associated with their natural frequency. The staff's position as stated in Section 3.9.2.II.2.a(2) of the SRP is that for equipment that can be modeled adequately as a one-degree-of-freedom system, only the use of a static load equivalent to the peak of the floor response spectra is acceptable. Either revise this paragraph to be consistent with the staff position, or provide the basis for the use of accelerations associated with the natural frequency.

b. The third paragraph in this section states that, for multi-degree-of-freedom systems, in lieu of using the peak acceleration value, the actual frequency may be calculated and the corresponding acceleration value may be used. It is not clear whether or not the 1.5 factor is also included in this corresponding acceleration value. Revise this paragraph to provide a clarification of this alternative.

c. The fifth paragraph in this section states that the equivalent static load method of analysis can also be used for small-bore piping. The staff's position as stated in Section 3.9.2.II.2.a(2) of the SRP is that an equivalent static load method is acceptable if justification is provided that the system can be realistically represented by a simple model and the method produces conservative results in terms of responses. Furthermore, Section 3.9.2.II.2.a (2) of the SRP states that the design and the associated simplified analysis account for the relative motion between support points and a factor of 1.5 is applied to the peak acceleration of the floor response spectrum. Alternatively, the use of a static load equivalent to the peak of the floor response spectra is acceptable for piping supported at only two points. Revise this paragraph to be consistent with the staff position, or provide the basis for the use of the equivalent static load method of analysis for small-bore piping.

210.49 Section 3.7.3.8.2.1 of the SSAR states that when run pipe is decoupled from the analytical model of the branch pipe, the connection point is considered to be anchored for seismic inertia analysis of the branch pipe. The response spectra for this analytical anchor are the spectra at the building floor location corresponding to run pipe supports near the connection point. The staff believes that the response spectra at the run pipe supports may not be a conservative assumption when compared with the actual configuration before decoupling. Revise this section to either change this assumption, or provide a more detailed basis for the assumption.

210.50 The last paragraph in Section 3.7.3.8.2.1 of the SSAR states that when the supporting system for auxiliary (branch) pipe is equipment, the supported pipe can be decoupled from the supporting equipment using the same criteria as when the supporting system is a piping system with the run pipe size replaced by the minimum dimension of the equipment. Describe how the minimum dimension (length, width, and height) of the equipment establishes an equivalent decoupling criteria to that of a run pipe.

210.51 Section 3.7.3.9 of the SSAR, "Combination of Support Responses," states that the effect of relative seismic anchor displacements are obtained by either using the worst combination of peak displacements or by proper representation of the relative phasing characteristics associated with different support inputs. Provide more details relative to how proper representation is obtained. Identify and justify any deviations from the guidelines in Section 3.9.2.II.2.g of the SRP. In addition, either revise or delete the exception to Section 3.9.2.II.2.g of the SRP in WCAP-13054, if applicable.

210.52 Section 3.9.1.1 of the SSAR, "Design Transients," discusses pressure, temperature, and flow transients, but does not include seismic events. The last paragraph in this section states that where applicable, in addition to the effects produced by the above transients, earthquake loadings must be considered, and references Section 3.9.3 for a description of how these loads are considered for the AP600. To be consistent with the guidelines in Section 3.9.1 of the SRP, Section 3.9.1 and Table 3.9-1 of the SSAR should include seismic events as one of these transients. Section IM, "Elimination of OBE," in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor Designs," dated April 2, 1993, presents the current staff position relative to accounting for earthquake cycles in fatigue analyses. Revise Section 3.9.1 and Table 3.9-1 to include the seismic events and the number of cycles consistent with this staff position.

210.53 Section 3.9.2.1 of the SSAR states that the preoperational piping vibration, thermal expansion, and dynamic effects tests will be conducted only on the first AP600 plant because standardization of piping design eliminates the need to test the response of piping to transients in subsequent plants. The discussions of these tests in Sections 14.2.8.1.78, 14.2.8.1.82, 14.2.8.2.18, and 14.2.8.2.20 are also limited to the first plant only. This is an unacceptable commitment. The purpose of these tests is to confirm that the applicable piping systems, restraints, components, and supports have been adequately designed, fabricated, and installed to withstand flow-induced dynamic loadings under the steady-state and operational transient conditions and to confirm that normal thermal motion is not restrained. The staff believes that one major cause of excessive vibration or excessive pipe movement can be attributed to improper support installation or loss of snubber functionality. Standardization of piping design will not provide assurance that such discrepancies do not exist. Therefore, the staff's position is that these preoperational tests are required to be conducted on all AP600 plants in accordance with the criteria discussed in Sections 3.9.2.1.1 and 3.9.2.1.2. Revise Sections 3.9.2.1, 14.2.8.1.78, 14.2.8.1.82, 14.2.8.2.18, and 14.2.8.2.20 to be consistent with the above position.

210.54 Section 3.9.2.1.1 of the SSAR, "Piping Vibration Details," states that if system vibration is evidenced during initial operation, the maximum amplitudes are measured and related to alternating stress intensity levels based on the guidance of ANSI/ASME OM, "Operation and Maintenance of Nuclear Power Plants," Part 3. However, the scope of this OM Standard is more broad than the brief discussion in this section. This standard provides general requirements for the assessment of vibration in all safety-related piping systems during preoperational and start-up testing. It includes steady state and transient vibration testing, acceptance criteria, and recommendations for corrective action when required. In addition, it provides guidance for the assessment of vibration levels of applicable piping systems during plant operation. For the preoperational piping vibration and dynamic effects tests on all AP600 plants, the staff's position is that a commitment to the full scope of ANSI/ASME OM, Part 3 should be provided. If exceptions are taken to any portion of this standard, they should be clearly delineated in the SSAR and the bases for such exceptions should be provided. Revise Section 3.9.2.1.1 to reflect this staff position.

210.55 To be acceptable, the discussions in Sections 3.9.2.1.2 and 14.2.8.2.18 of the SSAR require a more specific commitment to the preoperational piping thermal expansion test program procedures. The staff's position is that these tests should be conducted in accordance with the ASME OM Standard, Part 7, "Requirements for Thermal Expansion Testing of Nuclear Power Plant Piping Systems." This standard contains procedures to be used for the assessment of thermal expansion response and design verification of piping systems. Implementation of this standard ensures that the piping system is ready for testing and can expand and contract as required during all plant conditions. It verifies that (a) expected expansion can be accommodated by the piping system restraints, (b) movement is not obstructed by any unintentional restraints, and (c) response is within design tolerances. It also provides guidance for development of acceptance criteria, instrumentation, and measurement techniques, as well as corrective actions and methodologies for reconciling movements that differ from those specified by the acceptance criteria. Revise Sections 3.9.2.1.2 and 14.2.8.2.18 to provide a specific commitment that detailed test specifications for thermal expansion testing of piping systems during preoperational and start-up testing will be in full accordance with ASME OM Standard, Part 7.

210.56 The second paragraph in Section 3.9.2.1 of the SSAR states that the preoperational piping vibration, thermal expansion, and dynamic effects test programs will be conducted on ASME Class 1, 2, and 3, and other high energy piping systems. It is the staff's position that these test programs should include safety-related instrument sensing lines at least up to the first support in these lines. Revise Section 3.9.2.1 to provide a commitment to include such lines in these test programs.

210.57 In applicable Sections of Section 14.2 of the SSAR, provide commitments that the testing will be in accordance with the criteria in Sections 3.9.2.1.1 or 3.9.2.1.2.

210.58 Section 14.2.8.1.77 of the SSAR states that the reactor internals flow-induced vibration tests will be conducted on only the first AP600 plant. This is not consistent with the commitments in Section 3.9.2.4. All AP600 non-prototype plants are required to be tested in accordance with applicable guidelines in RG 1.20. Revise Section 14.2.8.1.77 to delete the commitment to test the first plant only and to include a reference to RG 1.20.

210.59 Sections 3.9.1.1 and 3.9.3.1.2 of the SSAR each contain the same brief discussion that states that the design of piping and component nozzles in the AP600 will minimize the potential for and the effects of thermal stratification and cycling. In Section 3.9.3.1.2, provide a description of the confirmation process to be implemented by the COL licensee to determine whether these effects have been minimized to an acceptable level. If this cannot be verified, describe the analyses and testing required to assure that the design has accounted for these effects, including the method and procedures necessary to define the stratified thermal profile.

210.60 Section 3.7 of the SSAR states that the operating basis earthquake (OBE) has been eliminated as a design requirement for the AP600. Section IM, "Elimination of OBE," in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor Designs," dated April 2, 1993 contains the staff's recommendations to the Commission relative to this issue. The staff has evaluated the impact of this proposal, and has identified the necessary changes to the current seismic design criteria and the appropriate technical actions necessary for Westinghouse to implement these changes for the AP600. The current staff positions relative to these changes are discussed in the attachment to this enclosure. Revise applicable portions of Section 3.9 of the SSAR to implement these positions.

210.61 Section 1.9.5.1 of the SSAR, "SECY-90-016 Issues," under <u>Intersystem LOCA</u>, states, in part, that the design pressure of the normal residual heat removal system (RNS) piping downstream of the RNS isolation valves (including RNS pump casings and heat exchanger tubes) is designed to an ultimate rupture strength equal to full reactor coolant system (RCS) operating pressure. Provide a more detailed description of the low pressure side design criteria in order to reduce the likelihood of an intersystem LOCA. Revise Section 1.9.5.1 and any other applicable sections of the SSAR to provide the following for all

AP600 systems where overpressurization of low-pressure piping systems due to RCS boundary isolation failure could result in rupture of the low-pressure piping outside containment:

a. Pipe and pipe fittings - Provide the pipe schedule number for all applicable diameters and materials.

b. Valves and flanges - Provide the American National Standard Class.

c. Pumps and heat exchangers - Provide the ratio of design pressure to RCS pressure.

210.62 To be consistent with Section 3.9.3 of the SRP, Appendix A, Table I, the staff's position is that Table 3.9.3-7 of the SSAR, "Minimum Design Loading Combinations for ASME Class 2 and 3 Piping," and Table 3.9.3-8, "Minimum Design Loading Combinations for Supports for ASME Class 1, 2, and 3, Piping and Components," should include SSE + DF in the loading combinations for the Level D condition for all Class 1, 2, and 3 components. Revise Table 3.9.3-7 to add this combination, and revise Table 3.9.3-8 to delete Note 3. Delete Note 7 to Table 3.9-6, if applicable. In addition, revise the exception to Section 3.9.3 of the SRP, Appendix A, Section C.1.2 in WCAP-13054, as applicable.

210.63 In Tables 3.9.3-5, 3.9.3-6, 3.9.3-7, and 3.9.3-8 of the SSAR, add a note to state that the method of combination of dynamic responses to loads is in accordance with the recommendations in NUREG-0484, "Methodology for Combining Dynamic Responses," Revision 1, dated May 1980. In addition, explain how Note 6 in Table 3.9.3-5 and Note 4 in Tables 3.9.3-6, 3.9.3-7, and 3.9.3-8 relate to these recommendations.

210.64 If an elastic-plastic method of analysis will be used in the design of any safety-related system, component, or support, identify each applicable item and revise either Section 3.9.1 or 3.9.3 of the SSAR to provide information consistent with the guidelines in Section 3.9.1.II.4 of the SRP.

210.65 Section 3.9.3.1.7 of the SSAR states that no special stress limits are required to provide functional capability of ASME Class 2 and 3 piping. The current staff position on this issue is documented in NUREG-1367, "Functional Capability of Piping Systems," dated November 1992. Revise applicable portions of the SSAR to commit to the positions in NUREG-1367 for all seismic Category I piping.

210.66 As stated in Section 3.9.3.2.2 of the SSAR, active valves are those whose operability is relied upon to perform a safety-related function during transients or events considered in the respective operating condition categories. This section references Tables 3.9-9 and 3.9-10 of the SSAR for stress limits used for active Class 1, and Class 2 and 3 valves, respectively. These tables provide no special stress limits for active valves. Note b in Table 3.9-9 states that valve operability is demonstrated by testing. The staff does not agree with the criteria for active valves that allows the calculated stresses to approach Service Level D limits. To provide further assurance of operability, in addition to testing, the staff's position is that the calculated maximum stress in the valves under all Service Levels be less than 1.10 times the allowable yield strength of the applicable material. This will help to insure that the deformation resulting from these loads will be small enough to allow operability. Revise Section 3.9.3.2.2 and Tables 3.9-9 and 3.9-10 of the SSAR to reflect this staff position.

210.67 The design and analysis requirements for ASME Class 1, 2, and 3 pressure-relieving devices discussed in Sections 3.9.3.3 and 10.3.2 of the SSAR do not appear to be consistent with staff positions on this issue. To be acceptable, such installations should be designed in accordance with ASME Section III, Appendix O, "Rules for the Design of Safety Valve Installations," as supplemented by the additional criteria in Section 3.9.3, Section II.2 of the SRP. Revise Sections 3.9.3.3 and 10.3.2 to be consistent with this position. In addition, delete the reference to ANSI/ASME B 31.1, Appendix 2 in Section 10.3.2.2.2.

210.68 Section 3.9.3.4 of the SSAR does not appear to specifically address allowable stress criteria for active component supports, where active is as defined in Section 3.9.3.2.2. The staff's position is that the stresses and associated deformations in such supports should be low enough to allow operability of the supported component. In Appendix 1A of the SSAR and Revision 1 to WCAP-13054 (under exceptions to Section 3.9.3 of the SRP), exceptions are taken to Position C8 in RG 1.124, and Paragraph B.5 in RG 1.30, which are the bases for the staff's position on this issue. The exceptions in Appendix 1A state that ASME Level C and D Service Limits are acceptable, however, when they are used, any significant deformation that might occur will be considered in the evaluation of equipment operability. Revise Section 3.9.3.4 to reference this exception and provide a more detailed discussion on how this significant deformation will be evaluated for the AP600 to meet the guidelines in Section 3.9.3.II.3 of the SRP. Appropriate revisions should also be made to Tables 3.9-9 and 3.9-10, and Appendix 1A of the SSAR, and to the exception to Section 3.9.2.II.3.a of the SRP in WCAP-13054.

210.69 Section 3.9.3.4.3 of the SSAR does not provide sufficient information for the staff to conclude that snubber operability will be assured. Revise this section to provide a more detailed discussion which incorporates the guidelines in Section 3.9.3.II.3 of the SRP. In addition, if applicable, provide a commitment to dynamically qualify all large bore hydraulic snubbers.

The discussion of Generic Safety Issue A-13, "Snubber Operability Assurance," in Section 1.9.4.2 of the SSAR should also be revised to reference the revised Section 3.9.3.4.3.

210.70 Section 3.9.5.2.4 of the SSAR states that the AP600 core barrel, upper and lower support plates, support columns, and radial key supports are considered core support structures and constructed to ASME Subsection NG. It further states that for other internal structures, Article NG-3000, "Design," does not specifically apply and that these other internals are designed and fabricated using the ASME Code as a guide. In Sheet 38 of Table 3.2-3 of the SSAR, all of the safety-related reactor internals are listed as AP600 Class C and the principal construction code is identified as ASME III, CS for all internals. This implies that all internals are constructed to ASME Subsection NG, which does not agree with the statement in Section 3.9.5.2.4 of the SSAR relative to other internals. If internal structures other that those identified as core support structures will not be constructed to NG-3000, revise this section to provide a more detailed description of the design criteria that is used for such items. Include a discussion describing how selected code rules and other requirements are used together to ensure structural adequacy and functionality of various internal structures at various conditions. In addition, revise Table 3.2-3 to be consistent with the revised Section 3.9.5.2.4.

210.71 Sheet 53 of Table 3.2-3 of the SSAR does not appear to contain the core barrel and the control rod drive mechanism (CRDM) housings as a part of the reactor system. Revise this table to include these two components. If the CRDM housings are considered a part of the reactor vessel, add a note to this effect.

210.72 Section 3.9.7.1 of the SSAR states that the shroud assembly and the CRDM seismic support plate, which are both part of the integrated head package (IHP), are required to provide seismic restraint for the CRDM and the valves and piping of the reactor head vent and are both classified as AP600 equipment Class D, seismic Category I. It further states that the shroud and seismic support plate are categorized as intervening elements using the rules of the ASME Code, Section III, Subsection NF, and are therefore not subject to the rules of NF. In addition, Section 3.9.7.3 of the SSAR states that these two components are designed to the guidelines of AISC-N690-1984. The staff does not agree with these classifications. Sections 3.9.4.3 and 5.4.12.1 of the SSAR state that the CRDM housing and the piping and equipment from the vessel head vent up to and including the second manual isolation valve, respectively, are ASME Class 1. In addition, the shroud is bolted to the ASME Class 1 reactor pressure vessel head. Therefore, the staff's position is that supports which provide seismic restraint for Class 1 components and are also attached to Class 1 components cannot be categorized as intervening elements as defined in ASME Subsection NF, but should be classified as AP600 Class C and constructed to all of the rules of Subsection NF. Revise Sections 3.9.7.1 and 3.9.7.3 and Sheet 38 of Table 3.2-3 to reflect this staff position.

210.73 The ASME Code requires that a design specification be prepared for all ASME Class 1, 2, and 3 components. The design specification is intended to become a principal document governing the design and construction of these components and should specify loading combinations and other design data inputs. The Code also requires a design report for all such components. In the past, as a part of its review of plants under construction, the staff reviewed documents related to design specifications and design reports for a small number of ASME Class 1, 2 and 3 pumps, valves, and piping systems. The staff intends to perform such a review for the first AP600 plant. The objective of this review will be to provide the staff with the basis for concluding that the AP600 design documentation meets the applicable requirements of ASME Section III, Subsection NCA. In the interim, either revise Section 3.9.3 of the SSAR, or submit a separate document referenced in the SSAR, to provide a detailed description of the procedures used for generating design specifications for procurement of ASME Class 1, 2 and 3 components. Include a specific commitment to state whether Westinghouse or the COL licensee will provide the final documentation for the staff's review.

210.74 In the SSAR, Appendix 1A takes exception to Position C.7.b in RG 1.124 relative to Service Level D allowable loads for ASME Class 1 linear-type component supports designed by the load rating method. Revision 1 to WCAP-13054 contains this same exception. The exception states that the AP600 will use rules in ASME Appendix F, Section F-1370(d). F-1370(d) has been replaced by the load rating rules in F-1332.7, which is acceptable to the staff. Section 3.9.3.4 of the SSAR states that all AP600 ASME Class 1 supports are designed to ASME III, Subsection NF and Appendix F. Revise the exceptions in Appendix 1A of the SSAR, WCAP-13054, and any other applicable SSAR section to reference F-1332.7 rather than F-1370(d).

210.75 In the SSAR, Appendix 1A takes exception to Position C.6.b in RG 1.130 relative to Service Level D allowable loads for ASME Class 1 plate-and-shell-type component supports designed by the load rating method. Revision 1 to WCAP-13054 contains this same exception. The exception presents an equation for an allowable load rating which is not consistent with ASME III, Appendix F, Section F-1332.7. Revise the exceptions in Appendix 1A of the SSAR, WCAP-13054, and any other applicable SSAR section to reference F-1332.7.

210.76 Section 3.6.2 of the SSAR does not appear to address the guidelines in Section B.1.c(4) of BTP MEB 3-1 in Section 3.6.2 of the SRP relative to structures that separate high-energy lines from essential components. Revision 1 to WCAP-13054 takes exception to this criteria and states that separating structures are designed for postulated terminal end breaks and high stress locations. This exception is not completely acceptable. The staff's position, as stated in Section 3.6.2 of the SRP, is that such structures should be designed to withstand the consequences of the pipe break on the high-energy line that produces the greatest effect on the structure irrespective of the fact that the pipe break criteria of Section 3.6.2 of the SRP might not require such a break location to be postulated. Revise Section 3.6.2 of the SSAR to add a commitment to this position, and delete the exception to this guideline in WCAP 13054.

210.77 The table in Revision 1 to WCAP-13054 that addresses Section 3.6.2 of the SRP lists MEB 3-1, Sections B.1.c.(5) and B.3.c.(4) as acceptable for the AP600 design. Both of these guidelines relate to qualifying equipment for environmental (temperature, pressure, and humidity) effects. Several portions of Section 3.6.2 of the SSAR briefly mention requirements for considering environmental effects. For example, Section 3.6.2.1.1.4 provides a commitment to evaluate leakage cracks in main steam and feedwater lines in the containment penetration area. However, Section 3.6.2 does not appear to contain any detailed discussion relative to the guidelines in the two MEB 3-1 sections. Revise Section 3.6.2 of the SSAR to include a commitment to these guidelines and provide a description of how environmental effects will be considered in the AP600 design of high and moderate energy piping systems.

210.78 Revision 1 to WCAP-13054 lists exceptions to Sections C.1.3.3(a) and C.1.3.3(b) of Appendix A to Section 3.9.3 of the SRP, that state that all pipe break loads are classified as Service Level D. The staff does not agree with these exceptions. Pipe breaks other than a LOCA or main-steam/feedwater pipe break are defined as design basis pipe breaks (DBPB) in Section 3.9.3 of the SRP and should be designed to Service Level C limits. Revise WCAP-13054 to delete these exceptions and revise Tables 3.9-5, 3.9-6, 3.9-7, and 3.9-8 of the SSAR to include Sustained Loads + DBPB under "Level C Service."

210.79 Revision 1 to WCAP-13054 lists an exception to Section C.1.3.4(a) of Appendix A to Section 3.9.3 of the SRP, that states that SSE loads are not combined with "non-LOCA" pipe ruptures. The staff does not agree with this exception. Design basis pipe breaks (DBPB) as defined in Section 3.9.3 of the SRP are non-LOCA loads and should be combined with SSE loads and designed to Service Level D limits. Revise WCAP-13054 to delete this exception and revise Tables 3.9-5, 3.9-6, 3.9-7, and 3.9-8 of the SSAR to include Sustained Loads + DBPB + SSE under "Level D Service."

210.80 Revision 1 to WCAP-13054 lists an exception to Section C.3.2 of Appendix A to Section 3.9.3 of the SSAR, that states that one-half SSE is evaluated to level C limits. This does not appear to be consistent with the staff's current position relative to the use of a single-earthquake design for the AP600. The attachment to Q210.60 contains this position. Revise the above exception to Section 3.9.3 of the SRP to be consistent with this staff position.

210.81 Revision 1 to WCAP 13054 lists an exception to Section 1 of Section 3.10 of the SRP, that states that safety-related equipment may be qualified, in part, based on properly documented experience data in accordance with Section 9.0 of IEEE 344-1987. As used in IEEE 344, experience data includes both seismic experience and previous qualifications. The staff has not accepted the use of seismic experience on either evolutionary or passive plants. In accordance with Revision 2 to RG 1.100, this method of qualification will be reviewed by the staff on a case-by-case basis. Revise the above exception and any other applicable SSAR section to reflect this staff position. In addition, include a statement in the SSAR that if dynamic qualification of seismic Category I electrical or mechanical equipment is accomplished by experience, the COL applicant should provide the following for NRC review and approval:

a. Identification of the specific equipment.

b. The details of the methodology and the corresponding experience data for each piece of equipment.

210.82 Revision 1 to WCAP-13054 lists an exception to Section 1.a.(1) of RG 3.10, that states that for electrical equipment, the only dynamic loads considered in testing are seismic loads, and that these seismic loads are not combined either by test or analysis with other dynamic loads. If there are any dynamic loads other than seismic that could affect either the equipment or the floor response for the equipment, these loads should be included in the equipment qualification either by test or analysis. Revise this exception to provide a detailed basis for not including such loads.

210.83 Revision 1 to WCAP-13054 lists an exception to Section 1.a.(2) of RG 3.10, that states that when performing seismic qualification of mechanical and electrical equipment by test, all accident loads are not superimposed on the seismic loads. Revise this exception to describe the types of accident loads that will not be superimposed on the seismic loads.

210.84 In the exception to Section 4 of Section 3.9.1 of the SRP described in Revision 1 to WCAP-13054, the last sentence states that a check valve which changes position in response to a pipe rupture event need not meet the criteria for active valves. This does not appear to agree with the staff position in Section B, "Definition of Passive Failure," of SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," dated March 28, 1994. The staff recommends that, for passive plant designs, check valves be redefined as active except for those whose proper function can be demonstrated and documented. Revise the exception in WCAP-13054, and any other applicable section in the SSAR to agree with this staff position.

210.85 Revision 1 to WCAP-13054 lists an exception to Section II.1.a(14)(b)iii of Section 3.10 of the SRP, that states that valve discs are not analyzed for pressure differential or impact energy resulting from a postulated pipe break, except for certain cases where a significant impact from a LOCA is expected. Either delete this exception, or in WCAP-13054 and Section 3.10 of the SSAR explain how this evaluation is made without an analysis to determine the design adequacy of the disc.

210.86 Revision 1 to WCAP-13054 lists Section 3 of Section 3.10 of the SRP as being not a part of the design process. Section 3 states that complete and auditable records of the seismic and dynamic qualification of equipment should be available and maintained by the applicant/licensee for the life of the plant. The staff agrees that for the AP600, such documentation will not be available for design certification. However, the staff's position is that the SSAR should state that the COL applicant should provide these records, which, in part, should consist of information similar to that in the sample equipment qualification data package in Attachment A of Appendix 3D of the SSAR. Revise the exception to Section 3.10 of the SRP in WCAP-13054, and Section 3.10 of the SSAR to provide this statement.

210.87 Revision 1 to WCAP-13054 lists an exception to Section 4 of Section 3.10 of the SRP, that states that some valves may exhibit an increase in leakage when subjected to seismic loading. The staff does not agree with this exception. As stated in Section 4 of Section 3.10 of the SRP, to satisfy GDC 14 and 30 in appendix A to 10 CFR Part 50, the qualification program for valves that are part of the reactor coolant pressure boundary should include testing or testing and analyses that demonstrate these valves will not experience any leakage, or increase in leakage, as a result of any loading or combination of loadings that the valves must by qualified for. Revise WCAP-13054 to delete this exception.

210.88 Revision 1 to WCAP-13054 lists an exception to Section 5c of Section 3.10 of the SRP, that states that Westinghouse does not prepare a Seismic Qualification Report (SQR), and that, in lieu of such a report, seismic qualification of equipment is documented in test reports, analysis reports, calculation notes, etc. contained in Westinghouse files. The staff's position is that an SQR should be prepared, and included in the documentation provided by the COL (see Q210.86). Revise this exception to state that the SQR should be submitted by the COL applicant.

In addition, verify the existence of design and analysis documentations of reactor internals, and provide a summary of the analysis results in conjunction with design limits.

210.89 Section 3.7.2.8 of the SSAR, "Interaction of Seismic Category II and Non-seismic Structures with Seismic Category I Structures" and Section 3.7.3.13 of the SSAR, "Interaction of Other Systems with Seismic Category I Systems" both address the criteria for protecting safety-related structures, systems, and components (SSCs) from adverse seismic interactions due to failure of non-safety-related SSCs (II/I). This is consistent with the staff's position for all types of nuclear plants up to and including the evolutionary designs. However, for the passive plants, the staff's position is that this criteria should not only be applicable to safety-related SSCs, but should also include those non-safety-related SSCs identified as important by the process to evaluate the issue of the regulatory treatment of non-safety related systems. Revise Sections 3.7.2.8 and 3.7.3.13 to reflect this position.

210.90 The criteria in Sections 3.7.2.8 and 3.7.3.13 of the SSAR, relative to protecting certain SSCs from adverse seismic interactions, are used for the design of the AP600. However, during the construction phase, interferences from field run items may lead to such interactions. To identify and correct such potentially adverse interactions, provide a statement in the SSAR that the COL applicant should describe the process for completion of the design of balance-of-plant and non-safety systems to minimize II/I interactions and propose procedures to be used for performing an assessment of the as-built plant to verify that the interaction of non-seismic SSCs with seismic SSCs does not affect the safety function of the seismic SSCs.

210.91 Revision 1 to WCAP-13054 lists an exception to Section C.4.1.(c) of Appendix A to Section 3.9.3 of the SRP, that implies that since the AP600 does not have an FSAR, information relative to how the criteria in Sections 1 and 2 of Appendix A have been implemented will not be in the SSAR. The staff's position is that this information should be available for the reactor system and most of the reactor coolant system, and should be discussed in the SSAR. Revise this exception in WCAP-13054 to identify those components/systems for which this information will be provided and revise the applicable SSAR section to include this information.

210.92 Revision 1 to WCAP-13054 lists an exception to Section 1.a.(6) of Section 3.10 of the SRP, that requires some clarification. It states that for an earthquake less than an SSE, each principle axis is simultaneously excited, and if no principal axis is evident, the equipment is positioned in the worst case orientation. Describe how the worst case orientation is determined if no principal axis is evident.

210.93 Revision 1 to WCAP-13054 lists an exception to Section 1.c of Section 3.10 of the SRP, that states that the AP600 design implements IEEE-323-1983 and IEEE-344-1987. The last sentence states that justification will be provided if the test sequence is not specifically followed, e.g., aging by analysis. The staff's position is that the last sentence should be deleted, and "exception" be replaced by "acceptable." If the last sentence is not deleted, describe how aging by analysis is accomplished.

210.94 Table 3.9-13 of the SSAR lists control rod drive mechanism (CRDM) production tests and their respective acceptance standard to ensure operational adequacy. In Section 3.9.4.4 of the SSAR, provide a discussion on how the functionality of the CRDM is ensured for seismic and LOCA loads.

210.95 Traditionally, the design of PWR internals is dominated by the LOCA loads due to postulating large breaks in the coolant loop. Thus, the internals have ample margins to resist an SSE, operation transients, and flow-induced vibrations, that generally induce less significant stress levels and deflections than that induced by the LOCA. Due to the application of leak-before-break (LBB), LOCA loads become less important in the internals design. In Section 3.9.2.5 of the SSAR, identify the largest LOCA used in the design of AP600 internals, and provide a discussion regarding how margins were maintained to ensure adequate defense of reactor internals against uncertainties of SSE and operational loads.

210.96 In Section 3.9.5 of the SSAR, discuss which part of the reactor internals (core support and other internals), and under what conditions, the criteria of Appendix F of the ASME Section III Code are applied. Since the Code does not ensure functionality, identify additional requirements used to ensure the safety function of internals.

210.97 In Section 3.9.5.3 of the SSAR, provide a more detailed discussion of the basis for the deflection allowables listed in Table 3.9-14.

210.98 During actuation of the passive core cooling systems, thermal stratification conditions are likely to exist. In an applicable section of the SSAR, describe how such conditions were considered in the design and analyses of the reactor vessel and reactor internals.

210.99 The cross sectional drawings shown in Figures 3.9-5 and 3.9-6 of the SSAR lack detailed descriptions. It is unclear how different parts of reactor internals are connected. In addition, identify the relative locations of the internals to each other and to the reactor vessel. Revise the figures to clarify the design detail shown.

210.100 The response to Q210.16 dated January 1, 1993 indicates that preoperational test data from several operating plants and from scale-model flow tests are used for the assessment of flow-induced vibrations of the AP600 reactor internals. The response also indicates that the assessment has not yet been finalized and the effort was planned to be completed in the first quarter of 1994. When this assessment is complete, revise Section 3.9.2 of the SSAR to provide a more detailed summary of the assessment results used (a) for verifying your conclusions on the adequacy of the AP600 reactor internals design to withstand flow induced vibration, and (b) to provide the basis for classifying the first AP600 plant as Non-Prototype, Category II in accordance with Position C.1.5 of RG 1.20.

210.101 Supplement the response to Q210.17 dated January 8, 1993 by providing key dimensions of the reactor vessel and supports and by specifying major design parameters (e.g., temperatures, pressure, etc.) of the reactor vessel and internals.

210.102 The response to Q210.18 dated January 14, 1993 indicates that the preoperational vibration test program for the initial AP600 plant remains to be developed. Thus, detailed information regarding the program, including types and locations of sensors to be installed, the bases used to establish expected and acceptable vibration levels, and the conditions at which data are to be acquired, is not available at this time. The staff's position is that such information is essential for ensuring design adequacy of reactor internals to withstand flow-induced vibrations under operational transients. Subsequent to the staff receiving an acceptable response to Q210.100, develop and provide such information in the SSAR for design certification review.

210.103 The response to Q210.19 dated January 8, 1993 identifies ASME Code criteria applicable to AP600 core support structures. Provide specific values of stress limits, deflection limits, and buckling stability limits for various core support structures. Also, provide the design limits of internal structures other than the designated core support structures.

210.104 In Revision 1 to WCAP-13054, under Section 3.9.2 of the SRP, an exception is taken to Position C.1 in RG 1.20, and Position C.2 in RG 1.20. These positions are listed as not applicable to the AP600 design certification because it applies to preoperational and initial startup testing. The staff cannot evaluate these issues until it receives acceptable responses to Q210.100 and Q210.102. Revise WCAP-13054 to agree with the resolution of those RAIs.

210.105 In Revision 1 to WCAP-13054, Section 4 in Section 3.9.2 of the SRP is listed as acceptable. However, in the "Comments/Summary of Exception" column, it states that the reactor internals for the first AP600 are classified as Non-Prototype Category II as defined in position C.1.5 of RG 1.20. The staff has not yet accepted this classification for the AP600. This decision will be made by the staff as a part of its review of the responses to Q210.100 and Q210.102. Revise WCAP-13054 to agree with the resolution of those RAIs.

210.106 The AP600 plant life design objective is 60 years. This proposed design life raises questions relative to the margins available in the current ASME Fatigue Design Curves. These margins were established almost 30 years ago and were obtained from best-fit curves of fatigue test data by applying a factor of either two on stress or twenty on cycles, whichever was more conservative at each point. These factors were originally intended to cover such effects as environment, size effect, and scatter of data. However, based on limited data currently available, the staff believes that these margins may not be sufficient to account for variations in the original fatigue test data due to various environmental effects. The ASME Code curves may not be revised for many years. Therefore, the staff's position is that until these curves are revised, all ALWR's should include a proposed approach for accounting for environmental effects in the fatigue analyses for all ASME Class 1 systems, components, and equipment and for the designs of all Class 2, and 3 systems, components, and equipment that are subjected to cyclic loadings, including operating vibration loads and thermal transient effects of a magnitude and/or duration so severe that the 60-year life cannot be assured by the required Code analyses. In Section 3.9.3 of the SSAR, provide a proposed approach to resolving this concern for the AP600.

210.107 In Section 3.9.3.4 of the SSAR, provide a commitment that for pipe support base plate designs, the applicable action items in IE Bulletin 79-02, Revision 2, dated November 8, 1979 will be met. The staff's position on this issue is as follows:

- If "undercut" type expansion anchor bolts will be used in the AP600, and, if the safety factors used for such bolts are different from those in IE Bulletin 79-02, provide the factors which will be used in the design of the "under-cut" type of expansion anchor bolt and the basis for these factors.

- Irrespective of the type of expansion anchor bolt that will be used, the staff requires a commitment to the action item in IE Bulletin 79-02 relative to pipe support base plate flexibility.

210.108 Section 5.2.1.1 of the SSAR, "Compliance with 10 CFR 50.55a," states that the Code of record for evaluations done to support the AP600 SSAR and the design certification is the 1989 Edition, 1989 Addenda of ASME Boiler and Pressure Vessel Code, Section III. At this time, 10 CFR 50.55a(b)(i) only endorses ASME Section III through the 1989 Edition. Revise Section 5.2.1.1 to delete the reference to the 1989 Addenda, or provide justification for its use.

210.109 In Section 5.2.1.2 of the SSAR, provide a list of the ASME Code Cases to be used in the AP600 plant design.

210.110 The response to Q210.25 dated January 22, 1993 and Section 3.9.6 of the SSAR both state that the AP600 inservice testing program will include safety-related ASME Class 1, 2, or 3 valves. The staff's position for passive plants, as recommended in Section H, "Inservice Testing of Pumps and Valves," of SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," dated March 28, 1994 is that those important non-safety-related pumps and valves identified by the regulatory treatment of non-safety systems (RTNSS) process should be designed to accommodate testing in accordance with ASME Code, Section XI. Specific positions on the inservice testing requirements for these components will be finalized when the staff completes its review of the RTNSS issue. Revise the response to Q210.25 and Section 3.9.6 of the SSAR to reflect this staff position.

ATTACHMENT FOR Q210.60

STAFF POSITION ON THE USE OF A SINGLE-EARTHQUAKE DESIGN FOR SYSTEMS, STRUCTURES, AND COMPONENTS IN THE AP600 STANDARD PLANT

A. INTRODUCTION

Appendix A to 10 CFR Part 100 requires, in part, that all structures, systems, and components of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public shall be designed to remain functional and within applicable stress and deformation limits when subject to an operating basis earthquake (OBE). Changes to Appendix A to Part 100 are being proposed to redefine the OBE to a level such that the function of the OBE can be satisfied without the need to perform explicit responses analyses.

The purpose of this staff position is to identify the necessary changes to existing seismic design criteria that are acceptable to the NRC staff for implementing the proposed rule change as it pertains to the design of safety-related systems, structures, and components in the Westinghouse simplified passive advanced light water reactor plant (AP600). These criteria apply only to the AP600 standard plant design and are not intended to replace the seismic design criteria approved by the Commission in the licensing bases of currently operating facilities. The guidelines provided herein are consistent with the Electric Power Research Institute (EPRI) Utility Requirements Document for passive plants.

B. BACKGROUND

In SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," dated January 12, 1990, the staff requested the Commission's approval to decouple the level of the OBE ground motion from that of the safe-shutdown earthquake (SSE). The Commission approved the staff's position in its staff requirements memorandum (SRM) of June 26, 1990.

In the draft Commission paper, "Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements," dated February 20, 1992, the staff further requested the Commission to approve eliminating the OBE from the design of systems, structures, and components in both evolutionary and passive advanced reactors designs. The proposed amendment to Appendix A of 10 CFR Part 100 would allow, as an option, that the OBE be eliminated from design certification when the OBE is established at less than or equal to one-third the SSE. In this manner, the OBE serves the function as an inspection level earthquake below which the effect on the health and safety of the public would be insignificant and above which the licensee would be required to shut down the plant and inspect for damage. The elimination of the OBE from design was requested by the EPRI and also recommended by the Advisory Committee on Reactor Safeguards (ACRS) in its letter of April 26, 1990.

In SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Passive Advanced Light Water Reactor Designs," dated April 2, 1993, the staff examined the safety impact of eliminating the OBE as it pertains to civil structures, piping systems, and equipment seismic qualification. Several recommendations were made by the staff to ensure that eliminating the OBE would not result in a significant decrease in the overall plant safety margin. The Commission approved the staff's recommendations in its SRM dated July 21, 1993. The following sections of this position paper contain the specific actions needed for the AP600 standard plant design to ensure that adequate safety margins are maintained when the OBE is eliminated from the design. The sections identify those actions needed for: (1) piping systems, (2) concrete and steel structures, (3) equipment seismic qualification, and (4) pre-earthquake planning and post-earthquake operator actions.

C. ASME CODE CLASS 1, 2, AND 3 COMPONENTS AND CORE SUPPORT STRUCTURES

The dynamic analysis methods to be used for seismic analyses of ASME Code Class 1, 2, and 3 components and core support structures in the AP600 shall use those methods described in the AP600 SSAR as approved by the NRC staff in its final safety evaluation report (FSER). The loads and load combinations to be used for evaluating ASME Code Class 1, 2, and 3 components and core support structures are provided in the AP600 SSAR and discussed in the staff's FSER. The OBE may be eliminated from the applicable design load combination when the following supplemental criteria are used.

1. Fatigue

In order to ensure adequate design considerations for the fatigue effects of earthquake cycles, it is necessary to establish, for a 60-year plant life, a bounding load definition and a number of earthquake cycles to account for the more frequent occurrences of lesser earthquakes and their aftershocks. For the AP600, an acceptable cyclic load basis for fatigue analysis of earthquake loading for ASME Code Class 1, 2, and 3 components and core support structures is two SSE events with 10 maximum stress cycles per event (20 full cycles of the maximum SSE stress range). This is approximately equivalent to the cyclic load basis of one SSE and five OBE events as currently recommended in the Standard Review Plan (NUREG-0800) Section 3.9.2. Alternatively, an equivalent number of fractional vibratory cycles to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Appendix D of IEEE Standard 344-1987.

2. Seismic Anchor Motion (SAM)

For the AP600, the effects of displacement-limited, seismic anchor motions (SAM) due to a safe-shutdown earthquake should be evaluated for safety-related ASME Code Class 1, 2, and 3 components and component supports to ensure their functionality during and following an SSE. The SAM effects should include (but are not limited to) relative displacements of piping between building floors and slabs, at equipment nozzles, at piping penetrations, and at connections of small-diameter piping to large-diameter piping.

For piping systems, the effects of seismic anchor motions due to a safe-shutdown earthquake should be combined with the effects of other normal operational loadings that might occur concurrently as specified in Section C.3.1 and C.3.2 of this position paper.

3. Piping Stress Limits

For ASME Code Class 1, 2, and 3 piping, the design requirements in the 1989 Edition of the ASME Boiler and Pressure Vessel Code, Section III, Subsections NB, NC, and ND shall be met. In addition, the following changes and additions to paragraphs NB-3650, NC-3650, and ND-3650 are necessary and shall be satisfied for piping systems when the OBE is eliminated from the design.

3.1 ASME Code Class 1 Piping Stress Limits

a. For primary stress evaluation (NB-3654.2), earthquake loads are not required to be evaluated for consideration of Level B Service Limits for Equation (9).

b. For satisfaction of primary plus secondary stress intensity range (NB-3653.1), in Equation (10), M_i shall be either (1) the resultant range of all loads considering one-half the range of the safe-shutdown earthquake or (2) the resultant range of moment due to the full range of the safe-shutdown earthquake alone, whichever is greater. The use of the safe-shutdown earthquake is intended to provide a bounding design for the cumulative effects of earthquakes of a lesser magnitude and is therefore to be included in consideration of Level B Service Limits for Equation (10). A reduced range (with an equivalent number of fractional vibratory peak cycles) of the safe-shutdown earthquake moment may be used for consideration of Level B Service Limits (but with a range not less than one-third of the maximum SSE moment range).

c. For satisfaction of peak stress intensity (NB-3653.2), the load sets developed in NB-3653.1 based on the above Position C.3.1(b) should be used in calculating the peak stress intensity, S_p, and the alternating stress intensity, S_{alt}, for evaluating the fatigue effects and cumulative damage.

d. For simplified elastic-plastic discontinuity analysis (NB-3653.6), if Equation (10) cannot be satisfied for all pairs of load sets, then the alternative analysis as described in NB-3653.6 should be followed. In addition, the following condition shall be satisfied:

$$S_{sam} = C_2 \frac{D_o}{2I} (M_i^* + M_i^{**}) \leq 6.0\, S_m \qquad \text{Equation (12a)}$$

where: S_{sam} is the nominal value of seismic anchor motion stress

M_i^* is the same as M_i^* in Equation (12)

M_i^{**} is the same as M_i in Equation (10), except that it includes only moments due to seismic anchor motion displacements caused by a safe-shutdown earthquake

The combined moment range ($M_i^* + M_i^{**}$) shall be either (1) the resultant range of thermal expansion and thermal anchor movements plus one-half the range of the safe-shutdown earthquake anchor motion or (2) the resultant range of moment due to the full range of the safe-shutdown earthquake anchor motion alone, whichever is greater.

3.2 ASME Code Class 2 and 3 Piping Stress Limits

a. For consideration of occasional loads (NC/ND-3653.1), earthquake loads (i.e., inertia and seismic anchor motion) are not required for satisfying Level B Service Limits for Equation (9).

b. For consideration of thermal expansion or secondary stresses (NC/ND-3653.2), M_c in Equation (10) is not required to include the moment effects of seismic anchor motions due to an earthquake.

c. For consideration of secondary stresses in Level D Service Limit (NC/ND-3655), the following condition should be satisfied:

$$S_s = i \frac{M_c^* + M_c}{Z} \leq 3.0\ Sh \qquad \text{Equation (10b)}$$

where: M_c^* is the range of moments due to seismic anchor motions due to a safe-shutdown earthquake

M_c is the range of moments due to thermal expansion

4. Pipe Break Postulation Without OBE

It is recognized that pipe rupture is a rare event which might only occur under unanticipated conditions, such as those which might be caused by possible design, construction, or operational errors; unanticipated loads or unanticipated corrosive environments. The staff's observation of actual piping failures have found that they generally occur at high stress and fatigue locations, such as at the terminal ends of a piping system at its connection to component nozzles. Currently, in accordance with Section 3.6.2 of the Standard Review Plan (NUREG-0800), Revision 2, dated June 1987, pipe breaks are postulated in high-energy piping at locations of high stress and high fatigue usage factor. The load combination used in calculating the high stress and usage factor includes normal and upset load conditions (i.e., pressure, weight, thermal, OBE, and other operational transient loadings).

From a historical viewpoint, the criteria for postulating high-energy breaks at specified locations were first introduced in the early 1970s. The basis for the mechanistic approach for selecting pipe break locations was derived from the premise that although pipe breaks could result from random events induced by unanticipated conditions, the failure mechanism and the expected location of failure would likely be caused by local conditions of high stress or high fatigue in the piping. In order to ensure that a sufficient number of pipe breaks would be postulated, breaks were recommended to be postulated for a wide spectrum of events to envelope the uncertainties of unanticipated failure mechanisms. Breaks were postulated at terminal ends of the piping, at high stress and high fatigue locations, and as a minimum at two additional intermediate locations when the stresses were below the high stress threshold limit. The resulting criteria that were incorporated in Section 3.6.2 of the SRP resulted in many postulated pipe break locations and caused the installation of numerous pipe rupture mitigation devices in nuclear plants.

In the mid-1980s, the NRC's Executive Director for Operations initiated a comprehensive review of nuclear power plant piping to identify areas where changes to the piping requirements could improve the licensing process as well as the safety and reliability of nuclear power plants. The NRC's Piping Review Committee (PRC), in an integrated effort with the nuclear industry under the Pressure Vessel Research Council, conducted a comprehensive study of piping criteria including the mechanistic pipe break postulation guidelines. The PRC found that when an excessive number of pipe rupture mitigation devices (i.e., pipe whip restraints and jet impingement shields) are installed on high-energy piping systems, the potential exists for piping systems to be overly constrained. This condition was found in several nuclear plants in which massive pipe restraints adversely affected the ability of the high temperature piping to freely expand during normal plant operation. The PRC also found through numerous dynamic tests and field observations of non-seismically designed piping systems that had undergone high seismic loadings that butt-welded piping possesses an inherent ability to withstand large seismic inertial loadings without failure.

As a result of the PRC's effort, the NRC staff recognized that the mechanistic pipe rupture criteria for selecting locations of pipe breaks resulted in an excessive number of pipe rupture mitigation devices that could hinder the normal operation of the plant and might not contribute significantly to the overall safety of the plant. Accordingly, the SRP was revised to reduce the number of postulated pipe breaks by (a) eliminating the need to postulate pipe breaks at the two arbitrary intermediate locations and (b) providing a leak-before-break approach in lieu of postulating pipe breaks when the system and material specific information is adequate to justify its application.

Based on recent dynamic pipe tests conducted by EPRI and the NRC, it has been demonstrated that the piping can withstand seismic inertial loadings higher than an SSE without rupturing. Thus, the staff believes the likelihood of a pipe break in a seismically-designed piping system due to an earthquake magnitude of one-third SSE is remote. Operating experience has shown that pipe breaks are more likely to occur under conditions caused by normal operation (e.g., erosion-corrosion, thermal constraint, fatigue, and operational transients).

On the basis of the above discussion, the staff concludes that no replacement earthquake loading should be used to establish postulated pipe rupture locations. Instead, the criteria for postulating pipe breaks in seismically-designed, high-energy piping systems should be based on factors attributed to normal and operational transients only. However, for establishing pipe breaks and leakage cracks due to fatigue effects, the calculation of the cumulative usage factor should continue to include seismic cyclic effects. The staff's revised criteria for pipe break postulation are provided below. The revised criteria are intended to ensure that breaks are postulated to occur at the most likely locations and to reduce the number of pipe rupture mitigation devices (e.g., pipe whip restraints and jet impingement shields) that might hinder plant operation without providing a compensatory level of safety.

The elimination of earthquake loads in the revised pipe break criteria below is justified, in part, on the assumption that the equipment environmental qualification and compartment pressurization analyses for the AP600 are based on a bounding load definition for each compartment. In addition, Westinghouse should state in the SSAR that the COL licensee should have a monitoring program for erosion-corrosion that provides assurances that procedures or administrative controls are in place to assure that the NUMARC program (or another equally effective program) is implemented and the structural integrity of all high-energy (two-phase as well as single-phase) carbon-steel systems is maintained as discussed in Generic Letter 89-08 and NUREG-1344, "Erosion/Corrosion-Induced Pipe Wall Thinning in U.S. Nuclear Power Plants," April 1989.

Consistent with the above staff positions, the guidelines provided in Branch Technical Position MEB 3-1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," of Section 3.6.2 of the SRP may be revised as follows:

B.1.b.(1).(a): Footnote 2 should read, "For those loads and conditions in which Level A and Level B stress limits have been specified in the Design Specification (excluding earthquake loads)."

B.1.b.(1).(d): "The maximum stress as calculated by the sum of Equations (9) and (10) in Paragraph NC-3652, ASME Code, Section III, considering those loads and conditions thereof for which level A and level B stress limits have been specified in the system's Design Specification (i.e., sustained loads, occasional loads, and thermal expansion) excluding earthquake loads should not exceed $0.8(1.8\ S_h + S_A)$."

D. DESIGN OF STRUCTURES

1. Use of RGs 1.143 and 1.27

The staff guidelines in RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," and in RG 1.27, "Ultimate Heat Sink for Nuclear Power Plants," provide for a seismic design of radwaste buildings and ultimate heat sink features based on the operating basis earthquake. With the elimination of the OBE, the staff finds that these structures and features should be demonstrated to withstand the safe-shutdown earthquake. The structural design criteria using the SSE loading should use the appropriate loads and load combinations provided in Section 3.8.4 of the SRP.

2. Seismic Instrumentation

Westinghouse should ensure that adequate design provisions allow for the placement of seismic instrumentation in the free field so that the control room operator can be immediately informed through the event indicators when the response spectra level and the cumulative absolute velocity (CAV) experienced at this location exceeds the shutdown level and can take the necessary actions. The details of the instrumentation requirements are discussed in Section F of this safety evaluation.

E. EQUIPMENT SEISMIC QUALIFICATION

When equipment qualification for seismic loadings is performed by analysis, testing, or a combination of both, the staff recommends the use of the IEEE Standard 344-1987 as endorsed in Regulatory Guide 1.100, Revision 2. This standard has detailed requirements for performing seismic qualification using five OBE events followed by an SSE event. With the elimination of the OBE, it is necessary to qualify equipment with the equivalent of five OBE events followed by one SSE event. Therefore, the staff concludes that equipment should be qualified with five ½SSE events followed by one full SSE event. Alternatively, a number of fractional peak cycles equivalent to the maximum peak cycles for five ½SSE events may be used in accordance with Appendix D of IEEE Standard 344-1987 when followed by one full SSE.

F. PRE-EARTHQUAKE PLANNING AND POST-EARTHQUAKE OPERATOR ACTIONS

The design certification of the AP600 using a single-earthquake (SSE) design is predicated on the adequacy of pre-earthquake planning and post-earthquake inspections for damage that are to be implemented by the COL applicant. The COL applicant shall submit to the NRC staff as a part of its application the procedures it plans to use for pre-earthquake planning and post-earthquake actions. For the AP600, the NRC staff finds the criteria developed by the Electric Power Research Institute (EPRI) in EPRI Reports EPRI NP-5930, EPRI NP-6695, and EPRI TR-100082 together with the amendments, additions, and changes outlined below for evaluating the need to shut down the plant following an earthquake to be acceptable.

EPRI NP-5930

The EPRI Report NP-5930 shall be used with the following exceptions:

1. A free field instrument must be used for determining the CAV and the spectral acceleration level.

2. The response spectrum check is as follows:

   The 5 percent damped ground response spectrum for the earthquake motion at the site exceeds (a) the corresponding OBE response spectral acceleration between 2 and 10 Hz, or it exceeds an acceleration of 0.20g between 2 and 10 Hz whichever is greater, or (b) the corresponding OBE response spectral velocity between 1 and 2 Hz or a velocity of 6 inches per second between 1 and 2 Hz, whichever is greater.

3. The licensee shall consider as sufficient evidence to shut down the plant the simultaneous exceedance of the 5 percent damped ground response spectrum enumerated in Item 2 and the CAV exceedance of 0.16 g-sec for any one frequency on any one component of the free field ground motion. The CAV shall be determined in accordance with EPRI Report EPRI NP-100082. Also, any evidence of significant damage observed during the plant walkdown in accordance with the EPRI Report NP-6695 recommendations shall be sufficient cause for plant shutdown.

4. The instrumentation installed at the nuclear power plant shall be capable of on-line digital recording of all three components of the ground motion and of converting the recorded (digital) signal into the standardized CAV and the 5 percent damped response spectrum. The digitizing rate of the time history of the ground motions shall be at least 200 samples per second and the band-width shall be at least from 0.20 Hz to 50 Hz. The pre-event memory of the instrument shall be sufficient to record the onset of the earthquake.

5. The system must be capable of routinely calibrating the response spectrum check of 0.20g. Also, the CAV of 0.16g-sec should be calibrated with a copy of the October, 1987 Whittier, California earthquake or an equivalent calibration record provided for this purpose by the manufacturer of the instrumentation. In the event that an actual earthquake has been recorded at the plant site, the above calibration shall be performed to demonstrate that the system was functioning properly at the time of the earthquake.

EPRI NP-6695

The EPRI Report NP-6695 shall be used with the following exceptions:

Section 3.1, Short-Term Actions

Item 3, "Evaluation of Ground Motion Records"

There is a time limitation of four hours within which the licensee shall determine if the shutdown criterion has been exceeded. After an earthquake has been recorded at the site, the licensee shall provide a response spectrum calibration record and CAV calibration record to demonstrate that the system was functioning properly.

Item 4, "Decision on Shutdown"

Exceedance of the EPRI criterion as amended by the NRC or observed evidence of significant damage as defined by EPRI NP-6695 shall constitute a condition for mandatory shutdown unless conditions prevent the licensee from accomplishing an orderly shutdown without jeopardizing the health and safety of the public.

Add item 7, "Documentation"

The licensee shall record the chronology of events and control room problems while the earthquake evaluation is in progress.

Section 4.3.4.1, "Safe Shutdown Equipment" (p. 4-7):

In addition to the safe shutdown systems on this list containment integrity must be maintained following an earthquake. Since the containment isolation valves may have malfunctioned during the earthquake, inspection of the containment isolation system is necessary to assure continued containment integrity.

Section 4.3.4, "Pre-Shutdown Inspection"

Exceeding the EPRI criterion or evidence of significant damage should constitute a condition for mandatory plant shutdown, as the staff stated in its recommendation for Section 3.1, item 4, "Decision on Shutdown."

G. CONCLUSIONS

Based on the changes to the existing seismic design criteria discussed above, the staff concludes that eliminating the operating basis earthquake from the design of systems, structures, and components in the AP600 standard plant will not reduce the level of safety provided in current regulatory guidelines for seismic design. On the contrary, the staff finds that the changes provide an enhancement to safety by refocusing current design requirements to emphasize those areas where failure modes are more likely to occur and by precluding the need for seismic design requirements that do not significantly contribute to the overall safety of the plant.

Contingent upon Westinghouse submitting a revision to its SSAR (including the appropriate changes to its Tier 1 Design Certification Material) reflecting the above criteria, the staff concludes that the elimination of the OBE from the design of systems, structures, and components, in the AP600 standard plant is acceptable.