May 20, 2003

Mr. Michael M. Corletti
Passive Plant Projects & Development
AP600 & AP1000 Projects
Westinghouse Electric Company
Post Office Box 355
Pittsburgh, Pennsylvania 15230-0355

SUBJECT: REQUESTS FOR ADDITIONAL INFORMATION - AP1000 DESIGN CERTIFICATION REVIEW (TAC NOS. MB5491 AND MB7247)

Dear Mr. Corletti:

By letter dated March 28, 2002, Westinghouse Electric Company (Westinghouse) submitted its application for final design approval and standard design certification for the AP1000.

The Nuclear Regulatory Commission (NRC) staff is performing a detailed review of your design certification application to ensure that the information is sufficiently complete to enable the NRC staff to reach a final conclusion on all safety questions associated with the design before the certification is granted.

The NRC staff has determined that additional information is necessary to continue the review. The requests for additional information (RAIs) are included in the enclosure. The topics covered in these RAIs include the areas of seismology and seismic design, hydrology and meteorology, materials application, quality assurance and reliability assurance program, initial test program, chemical technology, auxiliary systems, instrumentation and control, reactor systems, radiation protection, containment systems, and human systems interfaces. RAI 230.021 and 240.005 were provided to you during the structural audit on April 5, 2003. The remainder of the RAIs were sent to you via electronic mail on March 27, 31, April 1, 2, 4, 7, 9, 24, and May 2, 2003.

Please contact one of the following members of the AP1000 project management team if you have any questions or comments concerning this matter:

Mr. John Segala (Lead Project Manager) at (301) 415-1858, jps1@nrc.gov;
Mr. Joseph Colaccino at (301) 415-2752, jxc1@nrc.gov; or Ms. Joelle Starefos at (301) 415-8488, jls1@nrc.gov.

Sincerely,

/RA/

Joelle L. Starefos, Project Manager
New Reactor Licensing Project Office
Office of Nuclear Reactor Regulation

Docket No. 52-006

Enclosure: As stated

cc: See next page

Mr. John Segala (Lead Project Manager) at (301) 415-1858, jps1@nrc.gov;
Mr. Joseph Colaccino at (301) 415-2752, jxc1@nrc.gov; or Ms. Joelle Starefos at (301) 415-8488, jls1@nrc.gov.

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Requests for Additional Information (RAIs)
AP1000 Standard Design Certification

Series 230 - Seismology and Seismic Design

RAI 230.020

Section 3.7.2.3 - Procedure Used for Modeling:

In discussions with Westinghouse regarding the development of the Nuclear Island (NI) dynamic model, the Nuclear Regulatory Commission (NRC) staff identified instances in which this complex finite element model, which was developed by multiple organizations in different countries, has not produced acceptable results. The NRC staff is concerned as to the process used by Westinghouse to ensure the adequacy of the structural model. Specific examples where the model did not produce acceptable results include:

- During a public meeting in November 2002, the NRC staff requested that Westinghouse select a simple shear wall section from its model to compare the lateral deflection of the selected wall predicted by the computer analysis against the results of a hand calculation. The model results were not consistent with the hand calculation.

- The seismic analysis result of the Auxiliary and Shield Building shows net tension in the shield building wall. This suggests that during seismic excitation parts of the basemat will lift up from the rock surface resulting in changes in the basemat stresses.

During a conference call on January 21, 2003, Westinghouse agreed to inform the NRC staff of its intentions regarding how Westinghouse plans to address the issues of (1) peer review of its AP1000 design models; and (2) stiffness reduction of shear wall models. In a submittal dated March 13, 2003, Westinghouse provided its response. The response was not adequate for the following reasons:

a. Westinghouse has indicated its intention to conduct the peer review by a single expert who is already involved in the AP1000 design process. Although a peer review is not a requirement per the regulations, a review of the model to determine its adequacy by an individual who is associated with the development of the model does not appear to provide an independent review of the model.

b. Westinghouse stated that it has incorporated quality in its modeling and analysis of the NI in all of its activities conducted so far; however, the NRC identified that the seismic analysis result of the Auxiliary and Shield Building shows net tension in the shield building wall. This suggests that during seismic excitation parts of the basemat will lift up from the rock surface resulting in changes in the basemat stresses. This result does not suggest that the model is of sufficient quality.

c. Westinghouse has accepted the recommendation to adopt the criteria in Federal Emergency Management Agency (FEMA) documents for the stiffness of reinforced concrete shear wall structures. However, Westinghouse would only use it when performing new analysis. It claims that this effect will be covered by a peak broadening of +10 percent and -20 percent. The reduction in stiffness of shear walls has two effects: one, on the design of the structure itself, and two, on the structures, systems, and components (SSCs) supported by the structure. On the design of the structure, Westinghouse asserts that a reduction in frequency of about 7 percent will occur, based on some Japanese tests cited in NUREG/CR-6241. The NRC notes that the FEMA recommendations are most current, and are based on a scrutiny of a broad base of test results. Using the FEMA recommendation, the reduction in natural frequency can be as much as 60 percent of those calculated without the stiffness reduction. The respective order and the fundamental natural frequency change can lead to significant changes in the seismic load, hence the member forces. On the response of supported SSCs, the ordering of respective dominant frequencies and higher significant modes of response can result in unpredictable shapes of response spectra. Therefore, it is essential that the response spectra at several critical locations be developed and compared against those obtained from the original analysis using higher stiffness properties.

In light of the inadequacies cited above, Westinghouse should provide further information as to how the NI dynamic model and related calculations used for design certification (DC) satisfy the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A.

<u>RAI 230.021</u>

The results presented to address the liftoff concern during the April 3, 2003, public meeting did not include the potential effects of coupled horizontal and vertical ground motion input as well as the effect of potential uncertainty in the model parameters. Westinghouse needs to provide quantitative results to demonstrate that the peak dynamic toe pressure, the maximum subgrade pressure on the critical auxiliary building basemat slab panel and the response spectra at a few critical locations, are within the design values.

An acceptable method to address these concerns may be to conduct a dynamic time history analysis for the basemat uplift issue using a model that incorporates nonlinear subgrade springs (compression only for the vertical and release of horizontal springs where a loss of contact with the basemat occurs). The subgrade springs should incorporate the effects of uncertainty in foundation properties by varying them by plus 50 percent and minus 50 percent.

An acceptable approach to perform the dynamic time history analysis may be to use a rigid plate model with springs distributed uniformly throughout the basemat footprint using a lumped mass stick model representing the NI building containing the total inertia of the building. If this approach is used, the results of the nonlinear analysis should be used to compare the peak dynamic toe pressure, the maximum subgrade pressure on critical auxiliary building basemat slab panel and the response spectra at a few critical locations.

<u>RAI 230.022</u>

Analyses for Loads during Operation: The basemat is represented by a three-dimensional finite element model with the computer program ANSYS (Design Control Document (DCD) Reference 21). The model considers the interaction of the basemat with the overlying structures and with the soil. Provisions are made in the model for two possible uplifts. One is the uplift of the containment internal structures from the lower basemat. The other is the uplift of the basemat from the soil. The staff notes that uplift and slapping back of the containment internal structures on the basemat would affect both the seismic design loads and in-structure response spectra for all SSCs associated with the containment internal structure, and would also affect the seismic response of the steel containment shell. Please clarify how these effects have been addressed in the seismic analyses used for design of the containment and containment internal structures.

RAI 230.023

DCD Tier 2, Section 3.8.5.1 states that the foundation is built on a mud mat for ease of construction. The mud mat is lean, nonstructural concrete and rests on the load-bearing soil. Waterproofing requirements are described in DCD Tier 2, Section 3.4.1.1.1. The DCD does not contain adequate information to conclude that the nonstructural concrete mud mat can withstand the very high toe pressure predicted in the Westinghouse liftoff analysis. This would potentially affect the safety of the NI foundation mat under design basis combination of loads. Please discuss and propose revisions to the DCD to address this issue.

Series 240 - Hydrology and Meteorology

RAI 240.005

AP1000 DCD Section 2.5 used the AP600 as guidance. However, this section was not updated to incorporate the rulemaking for 10 CFR Part 100.23. The 1997 rulemaking revised the requirements for information to be provided by a future combined license (COL) applicant to support its site-specific information for demonstrating the suitability of the site. Westinghouse should revise DCD Section 2.5 to address the revised site-specific attributes.

Series 252 - Materials Application

RAI 252.010

In RAI 252.001, the staff requested information related to the geometry, fabrication, materials, accessibility for inspection, and operating conditions for control rod drive system penetrations, as motivated by recent operating experience, NRC Bulletins 2001-01, 2002-01, and 2002-02. Since the RAI was issued, the staff has issued Orders, EA-03-009, to operating license holders related to inspection for cracks in these penetrations. The staff subsequently issued follow-up questions to Westinghouse related to changes in design and fabrication to reduce residual stresses, ability to visually inspect 360 degrees around each nozzle, pre-service volumetric inspection, and determination of the operating head temperature. Westinghouse responded to the follow-up questions in a letter dated April 7, 2003.

The NRC staff believes that new inspection, test, analyses, and acceptance criteria (ITAAC) should be included in the DCD to address the issues discussed in your RAI responses. Please provide proposed ITAAC related to the issues noted above.

RAI 252.011

Operating experience continues to show cracking of Alloy 600 components. Recent experience appears to indicate that cracking has even occurred in welds or components not previously expected to crack based on the temperature of the weld or component and the time in service. The staff believes that the use of Alloy 690 materials in contact with the reactor coolant is a substantial improvement over the use of other materials currently in wide use in the industry. However, data is not presently available to demonstrate that cracking in these welds and components will not occur over the projected 40-year COL period of an AP1000 plant. The staff also believes that bare metal visual inspection of these locations is highly effective in identifying locations where cracking occurs.

a. Please provide information to describe the extent to which the insulation of all Alloy 600/690 components and welds in the reactor coolant pressure (RCP) boundary (not just upper reactor vessel head penetrations) will be designed to readily facilitate bare metal visual inspection during refueling outage conditions.

b. Please provide proposed ITAAC to verify that all Alloy 600/690 components and welds in the RCP boundary are identified and are readily accessible for bare metal visual inspection.

RAI 252.012

The staff reviewed DCD Tier 2 Section 5.3.4 as it applies to pressurized thermal shock in accordance with SRP 5.3.2, “Pressure-Temperature Limits and Pressurized Thermal Shock.” Section 50.61 of 10 CFR Part 50, “Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events,” defines the fracture toughness requirements for protection against pressurized thermal shock (PTS) events. Section 50.61 establishes the PTS screening criteria, below which no additional action is required for protection from PTS events. The screening criteria are given in terms of reference temperature (RT_{PTS}). These criteria are 148.0°C (300°F) for circumferential welds and 132.2°C (270° F) for plates, forgings, and axial welds. To verify that the design will be in accordance with the regulatory requirements associated with PTS, the applicant needs to provide an appropriate ITAAC. The following is a suggested Design Commitment for this ITAAC: The amount of copper and nickel in the reactor vessel materials and the projected neutron fluences for the 40 year period of the COL will result in RT_{PTS} values lower than the screening criteria contained in 10 CFR 50.61.

Series 260 - Quality Assurance and Reliability Assurance Program

RAI 260.004

In the AP1000 DCD, Section 17.4.1, Revision 0, Westinghouse stated that the design reliability assurance program (D-RAP), as shown in Figure 17.4-1, is implemented in three phases. The first phase, the phase, defines the overall structure of the AP1000 D-RAP, and implements those aspects of the program which are applicable to the design process. During this phase, risk-significant SSCs are identified for inclusion in the program using probabilistic, deterministic, and other methods. Phase II, the post- process, develops component maintenance recommendations for the plant’s operations and maintenance activities for identified SSCs. The third phase is the site-specific phase, which introduces the plant's site-specific SSCs to the D-RAP process. Phases I and II are performed by the designer. Phase III is the responsibility of the COL applicant.

The NRC determined that it is not acceptable for Westinghouse to complete Phase II following issuance of a design certification (DC) for the AP1000 design. Current NRC staff policy does not allow the applicant to carry open items to the post DC phase. In accordance with the acceptance criteria for Standard Review Plan (SRP) Section 17.4, Reliability Assurance Program (RAP), Westinghouse should not have post DC issues in the DCD for the AP1000 design. Westinghouse may complete this activity prior to issuance of a DC or the COL applicant may complete this activity. Clarification of this Phase II activity is necessary.

<u>RAI 260.005</u>

c. In DCD Section 17.4.7.2.1, Information Available to COL Applicant, Revision 0, Westinghouse states:

   To support the COL applicant's D-RAP Phase III and O-RAP, the following information is provided:

   a. The list of risk-significant SSCs identified during the design phase
   b. The PRA (Probabilistic Risk Assessment) assumptions for component unavailability and failure data
   c. The analyses performed for components identified as major contributors to total risk, with the dominant failure modes identified and prioritized. The suggested means for prevention and mitigation of these failure modes forms the basis for the plant surveillance, testing, and maintenance programs.

The NRC staff cannot find some of this information in Table 17.4-1. The staff is requesting that Westinghouse add appropriate cross-references to DCD Section 17.4.7.2.1 for Items 2 and 3 noted above. In addition, the staff is requesting that Westinghouse add the appropriate cross-references in the AP1000 DCD, or PRA, to DCD Section 17.4.7.2.1 on component unavailability and failure data and the dominant failure modes for SSCs included in the D-RAP.

b. In DCD Section 17.4.7.4, Westinghouse provided an example of D-RAP implementation with the automatic depressurization system (ADS) for a selection of components that are in the D-RAP for the AP1000 design. In DCD Section 17.4.7.4, Westinghouse states that "the design and analytical results presented here are intended as an example and do not reflect the current AP1000 design." The NRC staff determined that this wording is confusing.

<u>RAI 260.006</u>

The NRC staff reviewed DCD Section 1, Appendix 1A, "Conformance with Regulatory Guides," and finds that Westinghouse has taken exceptions to certain quality assurance (QA) implementation guidance in the following Regulatory Guides (RGs). Specifically, RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants;" RG 1.38, "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water-Cooled Nuclear

Power Plants;" and RG 1.39, "Housekeeping Requirements for Water-Cooled Nuclear Power Plants."

The RGs reference use of American National Standards Institute (ANSI) standards N45.2-1, N45.2-2, and N45.2-3. The ANSI standards are now updated and incorporated into American Society of Mechanical Engineers (ASME) NQA-1 and NQA-2. The updates in ASME NQA-1 and NQA-2 are compatible to the ANSI standards with some new implementation guidance; therefore, the NRC staff finds that these exceptions to the RGs are acceptable. However, similar to RG 1.39, Westinghouse should add the following statement in DCD Appendix 1A to the exception taken in RG 1.37 and 1.38: "See Section 17.5 for Combined License information items."

<u>RAI 260.007</u>

As noted above in DCD Section 1, Appendix 1A, Westinghouse took exception to RG 1.28. Westinghouse states, in part, that "Section 2, Quality records requires programmatic nonpermanent records to be retained for 3 years. An additional requirement states that programmatic records shall be retained at least until the date of issuance of the full power operating license of the unit. A definitive schedule for obtaining a full power operating license does not exist. Westinghouse will follow a records retention plan that is keyed to the Final Design Approval. Compliance will be accomplished by initiating a retention period of 3 years from programmatic records starting on the date that NRC issues a AP1000 FDA."

RG 1.28, Regulatory Position C.2, Quality Assurance Records, states, in part, that programmatic nonpermanent records should be retained for at least 3 years. For programmatic nonpermanent records, the retention period should be considered to begin upon completion of the activity. In addition, product and programmatic nonpermanent records should be retained at least until the date of issuance of the full power operating license of the unit.

Since RG 1.28 states, in part, that programmatic nonpermanent records should be retained at least until the date of issuance of the full power operating license of a unit, the NRC staff could not determine if compliance with RG 1.28 would be achieved since programmatic nonpermanent will only be retained for 3 years following a final design approval (FDA). 10 CFR Part 52.55, "Duration of Certification," states, in part, that a standard design is valid for 15 years from the date of issuance. RG 1.28, Table 1, includes design and procurement programmatic nonpermanent records that can be discarded before a COL purchases the AP1000 design. Westinghouse should provide a list of the specific record types they are proposing to discard after three years. Westinghouse should also provide additional justification for discarding each of these record types after FDA.

<u>RAI 260.008</u>

The NRC staff also noted that in DCD Section 17.6, References, did not list some documents discussed in DCD Section 17.3:

- Westinghouse Electric Company Quality Management System (QMS), Revision 5, dated October 1, 2002.

- WCAP-15985, AP1000 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process, Revision 1, dated April 2003

Westinghouse should add these references to DCD Section 17.6. In addition, there is no reference to a project specific quality plan for the AP1000 design similar to Reference 4, WCAP-12600, Revision 4, "AP600 Advanced Light Water Reactor Design Quality Assurance Program Plan," January 1998.

Series 261 - Initial Test Program

RAI 261.014

In the exception to RG 1.41, Westinghouse states:

> The guidelines are followed for Class 1E dc power systems during the preoperational testing of the AP1000 redundant onsite electric power systems to verify proper load group assignments, except as follows. Complete preoperational testing of the startup, sequence loading, and functional performance of the load groups is performed where practical. In those cases where it is not practical to perform complete functional testing, an evaluation is used to supplement the testing.

Based on this exception to RG 1.41, the NRC staff requests the following specific information:

a. Which Regulatory Position in RG 1.41 does this exception apply to?

b. Which system (ac, dc, or both) does this exception apply to?

c. Specify the "cases" where it is not practical to perform functional testing of onsite power supplies and for each case justify why an evaluation process is an adequate substitute for preoperational testing?

RAI 261.015

In the exception to RG 1.68, Appendix A, Item 4.t, Westinghouse states:

> For the AP1000, natural circulation heat removal to cold conditions using the steam generators is not safety-related, as in current plants. This safety function is performed by the PRHR [passive residual heat removal]. Natural circulation heat removal via the PRHR is tested for every plant during hot functional testing. Therefore, Westinghouse has met the intent of the previous licensing commitments for natural circulation testing.

The NRC found that the exception to RG 1.68, Appendix A, Item 4.t, contradicts the low power tests in DCD test abstracts 14.2.10.3.6, "Natural Circulation (First Plant Only)" and 14.2.10.3.7, "Passive Residual Heat Removal Heat Exchanger (First Plant Only)." The exception to RG 1.68 states, in part, that "the PRHR is tested for every plant during hot functional testing." The lower power test abstracts are performed on the first plant only. DCD Section 14.2.10.3.7, also states, in part, "Also note that this test is not required to be performed if a large scale test of the AP600 or AP1000 type passive residual heat removal heat exchanger has been conducted, and has provide[d] data confirming adequate heat removal capability." The NRC staff requests the following information:

a. Westinghouse should clarify and justify the inconsistent natural circulation testing requirements in the exception to RG 1.68 and in test abstracts 14.2.10.3.6 and 14.2.10.3.7.

b. Should natural circulation testing be performed on every plant, the first plant only, or is it not required if a large scale test facility provides data confirming the adequacy of heat removal capability?

RAI 261.016

In the response to RAI 261.007b, Item 2, Westinghouse stated that the pseudo rod ejection test is performed as part of the rod cluster control assembly out of bank measurements in DCD subsection 14.2.10.4.6. Westinghouse notes that this test is only performed on the first plant to validate the analysis.

The NRC staff determined that the pseudo rod or Rod Cluster Control Assembly (RCCA) ejection test is performed in test abstract 14.2.10.4.6; therefore, RAI 261.007b, item 2 is partially resolved. However, Westinghouse states that this test is performed on the first plant only. The NRC staff determined that Westinghouse should clarify whether this test should be performed for every AP1000 plant or justify that this test is a first-plant-only test as described in DCD Section 14.2.5. The NRC staff also notes that DCD section 14.4.6 requires the COL applicant or licensee to either perform the tests listed in DCD subsection 14.2.5 or provide justification that the results of the first-plant-only tests are applicable to subsequent plants.

RAI 261.017

In the response to RAI 261.007b, Item 3, Westinghouse states that the rod cluster control assembly out-of-bank measurements test is not performed at full power as it would cause the plant to exceed peak power limits.

The NRC staff notes that in RG 1.68, Appendix A, Section 5, item i states, in part, demonstrate the capability and/or sensitivity, as appropriate for the facility design of incore and excore neutron flux instrumentation, to detect a control rod misalignment equal to or less than the technical specification (TS) limits (50 percent, 100 percent). Although the NRC staff agrees that this test should not be performed at a power level that could cause the plant to exceed thermal limits, the test should be performed at power levels consistent with RG 1.68. Westinghouse should either perform the test at a higher power level consistent with RG 1.68 or provide additional information to justify performing this test at a maximum of 50 percent power.

RAI 261.018

In the response to RAI 261.007b, Item 5, Westinghouse states that the dynamic response of the plant to close all main steam isolation valves (MSIVs) is bounded by a plant trip from 100 percent power, which is performed in test abstract 14.2.10.4.24. MSIV testing is performed during preoperational testing to avoid an at-power transient. This is consistent with the valve in-service test requirements (DCD Table 3.9-16). Please see note 20 of DCD Table 3.9-16.

The NRC staff lacks sufficient information to conclude that the plant trip from100 percent power is bounded for the MSIV closure transient. The NRC staff requests additional information to provide the basis for the statement that the MSIV closure transient is bounded by a plant trip from 100 percent power.

Series 281 - Chemical Technology

RAI 281.004

Westinghouse has specified that 27,540 pounds of trisodium phosphate (TSP) stored in the basket in the containment sump should be sufficient to provide enough buffering action for maintaining pH in the containment sump water after a loss-of-coolant accident (LOCA) in the range between 7 and 9.5. This will minimize formation of elemental iodine in the sump water. The staff would like to know how the amount of TSP needed for controlling pH in the sump water was determined. Please provide the following information:

a. Sources of boric acid solutions introduced into the containment sump after a LOCA and volume and concentration of the resultant solution.

b. Any other sources (including identified and unidentified reactor coolant system leakage) and amounts of acidic and basic chemical species which were taken into consideration in determination of the amount of TSP.

c. Description of the procedure used for determining the amount of TSP. Describe it and specify whether it is based on theoretical considerations or experimental data.

Series 410 - Auxiliary Systems

RAI 410.022

Several pieces of HVAC equipment are identified in the ITAAC, however no corresponding reference in the DCD can be found for these pieces of equipment. The specific pieces of equipment in question are identified below:

Tag No.	Equipment
VBS-MA-11 and MA-12	Instrumentation and Control (I&C) Divisions B and C Ancillary Fans
VXS-MS-04A through D	MSIV Compartments A, B C & D Air Handling Units
VXS-MS-08A & B	Valve Piping Penetration Rooms A & B Air Handling Units
VXS-MY-W01A, B & C	Annex Building Nonradioactive Equipment Room Unit Heaters
VXS-MS-07A and B	Mechanical Equipment Area Air Handling Units
VAS-030	Fuel Handling Area Differential Pressure Indicator
VAS-032	Annex Building Differential Pressure Indicator
VAS-033	Auxiliary Building Differential Pressure Indicator

Please revise the DCD accordingly.

Series 420 - Instrumentation and Control

RAI 420.047

During a public meeting on March 5, 2003, at the Westinghouse Automation Headquarters in Monroeville, Pennsylvania, Westinghouse gave formal presentations regarding the AP1000 I&C design overview and the status of Westinghouse's Common Qualified Platform. The NRC staff asked the following questions regarding the material which Westinghouse presented:

a. In light of the reactor vessel head degradation at the Davis-Besse Nuclear Plant, does the AP1000 design utilize special monitoring devices (such as sensors) for the identification of small leaks?

b. Provide justification as to why the Steam Dump Block, as described in DCD Chapter 7, is not listed in the AP1000 TSs.

c. Does the DCD state that the remote shutdown workstation has indication on all of the time? If not, then provide justification as to why it is acceptable to not have indication on all of the time.

RAI 420.048

The staff is reviewing the adequacy of the AP1000 communication system, section 9.5.2 and associated ITAAC's in the Design Control Document. The staff requests answers to the following questions in order to complete its review.

a. 10 CFR 73.55 (e) and (f) discusses that placement of backup power supplies for certain communication systems be in vital areas. This has not been mentioned in the DCD.

b. 10 CFR 73.55 (g) mentions testing requirements for certain communication systems. This has not been mentioned in the DCD.

c. Section 9.5.2 of the NRC's SRP, NUREG-0800 provides reviewer guidance on the design of communication systems (i.e., inter-plant and plant to offsite). Part of that guidance follows for wireless systems:

   Communications system will be protected from EMI/RFI effects of other plant equipment and there will be adequate testing and field measurements where necessary to demonstrate effective communications.

Section 9.5.2 of the NRC's SRP discusses the general requirement that addresses the need for communication equipment to provide effective communication during the "full spectrum of ...conditions ...under maximum potential noise levels."

Please discuss communication testing for plant startup and operations in sufficient detail to allow the staff to understand how effective communications will be demonstrated including EMI/RFI effects on the equipment. Please include how effective communications will be sustained for maximum potential noise levels as described above.

d. Have ITAAC's been identified for the Communication system (EFS) as discussed in 9.5.2 beyond those given in table 2.3-19, EFS and 3.1-1 Technical Support Center/Operations Support Center (TSC/OSC). If not, how will there be assurance that the appropriate tests and confirmatory criteria have been accomplished to meet requirements of 10 CFR, and noise level considerations for worse case postulated noise levels.

Series 440 - Reactor Systems

RAI 440.188

Generic Letter (GL) 93-04 identified a potential problem with rod control system failure and inadvertent withdrawal of a single rod control cluster assembly. WCAP-13854, Revision 1, "Ron Control System Evaluation Program," provided Westinghouse Owners Group's response to the issues raised in GL 93-04. WCAP-15800, Revision 1, "Operational Assessment for AP1000," indicates that the resolution of GL 93-04 for the AP1000 design is addressed in the AP1000 DCD Section 3.9.4. The staff reviewed Section 3.9.4 and could not find the description of the resolution of GL 93-04.

a. Please identify where, in DCD Section 3.9.4, the GL 93-04 issue is addressed, and/or describe how the issue is resolved for the AP1000 design, including the design and surveillance tests of the AP1000 rod control system which are consistent with WCAP-13854 and is acceptable for the resolution of GL 93-04.

RAI 440.189

For your long-term cooling analysis described in DCD Section 15.6.5.4C, Figures 15.6.5.4C-16-17 allocated only two nodes for the core. Please provide justification that no adiabatic heating takes place in the core during long-term cooling.

Series 471 - Radiation Protection

RAI 471.011

RG 8.12 (regarding criticality monitors) has been withdrawn. References to this RG were deleted in Section 1.9.1.4 of the DCD but not in Section 11.5.6. DCD Section 11.5.6 also references 10 CFR 70.24. Since the issuance of the AP600 safety evaluation, there has been new guidance on criticality monitors (namely 10 CFR 50.68). Please provide the applicable guidance that the AP1000 design is using for criticality monitors (i.e., are you still complying with 10 CFR 70.24 or 50.68?).

DCD Section 12.1.3, references RG 8.3, which has been withdrawn.

RG 8.13 (Instruction Concerning Prenatal Radiation Exposure) is referenced in the 1981 version of the SRP and should have been referenced in the COL Action Item in Section 12.1.3 of the DCD along with the other RGs.

Series 480 - Containment Systems

RAI 480.010

DCD Tier 2, Chapter 16, contains proposed TS for AP1000 plants. In Section 5.5, "Programs and Manuals," is TS 5.5.8, "Containment Leakage Rate Testing Program." It includes this passage:

> The peak calculated containment internal pressure for the design basis LOCA, P_a, is less than the design pressure of containment.

In contrast, the WOG Standard TS states:

> The calculated peak containment internal pressure for the design basis LOCA, P_a, is [45 psig].

Option B of Appendix J to 10 CFR Part 50 requires the numerical value of P_a to be specified in the TS.* The AP600 proposed TS did so, correctly, but this has been changed for AP1000, without explanation. Please provide justification for why the AP1000 DCD does not comply with the regulations.

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* Appendix J allows any plant to choose to conform to either Option A of Appendix J (Prescriptive Requirements), Option B (Performance-Based Requirements), or a specific combination of Options A and B. Their TS must specify which choice they have made. The WOG STS contains three versions of this TS, to account for these possibilities. Two of the versions (Option B, and Options A and B combined) specify the value of P_a, but the Option A version does not. This is because Option A does not require it; Option B does. The AP1000 DCD allows COL applicants to choose which option of App. J they want, but it is highly unlikely that an applicant will choose Option A alone. All operating plants today have chosen either Option B or a combination of Options A and B, because of the millions of dollars saved by using Option B. Also, the AP1000 proposed TS follow the Option B model.

RAI 480.011

DCD Tier 2 Section 6.2.5, "Containment Leak Rate Test System," has a bad reference. The section cites Appendix J to 10 CFR Part 50, as it should, but refers the reader to Reference 14, which reads as follows:

> 14. 10 CFR 50, Appendix J (Draft Proposed Revision), "Containment Leak Rate Testing," January 10, 1992.

The Draft Proposed Revision should not be referenced in the DCD for the following reasons:

- The cited document is not generally available to the public. It was not published in the *Federal Register*. It was not finalized. Even the NRC would have difficulty finding this document in its files.

- This document predates the effort to revise Appendix J that culminated in the issuance of Option B of the regulation in 1995. Thus, the cited document would have none of the Option B information which was clearly used by Westinghouse to prepare the DCD.

- Appendix J has not changed since 1995, so there is no reason to cite anything other than the current regulation.

It seems clear by reading the DCD that the AP1000 design uses the current regulation. This appears to simply be a left-over, out-of-date reference. Please remove the reference to the "Draft Proposed Revision" or provide the basis for using the draft proposed revision.

Series 620 - Human Systems Interfaces

RAI 620.044

Section 18.3, "Operating Experience Review" (OER), Criterion 5, "Risk-Important Human Actions." The OER should identify risk-important human actions that have been identified as different or where errors have occurred. The human actions should be identified as requiring special attention during the design process to lessen their probability. Neither the DCD, Revision 3, nor WCAP-14645, Revision 2, address how the OER covers this NUREG-0711 item. Please discuss how the DCD describes this information.

AP 1000

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