September 30, 2002

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

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION LETTER NO. 12 - AP1000 DESIGN CERTIFICATION REVIEW (TAC NO. MB4683)

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 as Enclosure 1. The topics covered in RAIs include the areas of fire protection, auxiliary systems, effluent treatment, containment systems, and inspections, tests, analyses, and acceptance criteria (ITAAC). These RAIs were sent to you via electronic mail on September 20, 24, and 27, 2002. You agreed that Westinghouse would submit a response to these RAIs by December 2, 2002. Receipt of the information by December 2, 2002, will support the schedule documented in our letter dated July 12, 2002.

Enclosure 2 contains a history of previously-issued RAI correspondence.

If you have any questions or comments concerning this matter, you may contact me at (301) 415-3053 or ljb@nrc.gov.

Sincerely,

/RA/

Lawrence J. Burkhart, AP1000 Project Manager
New Reactor Licensing Project Office
Office of Nuclear Reactor Regulation

Docket No. 52-006

Enclosure: As stated

cc: See next page

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Requests for Additional Information (RAIs)
AP1000 Standard Design Certification
Series 280 - Fire Protection
Series 410 - Auxiliary Systems
Series 460 - Effluent Treatment
Series 480 - Containment Systems
Series 640 - Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

Series 280 - Fire Protection

280.001

Section 9.5.1.2.1.1 and Item 55 of Table 9.5.1-1 of the AP1000 Design Control Document (DCD) notes that the stairwells outside of primary containment serving as escape routes, access for firefighting, or access routes to areas containing equipment necessary for safe shutdown have not been enclosed in masonry or concrete towers with a minimum fire rating of 2 hours as specified in Position C.5.a.6. of CMEB 9.5.1 of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports." This includes fire areas: 1201 AF 01, 1202 AF 01, 1202 AF 05, 1204 AF 02, 1205 AF 01, 2000 AF 02, 2009 AF 01, 2003 AF 02, 4001 AF 01, 4001 AF 01, 4002 AF 02, and 4003 AF 02. The staff previously granted Deviation 9.5.1-2 for the use of gypsum stair towers in lieu of concrete or masonry for the AP600 in NUREG-1512, "Final Safety Analysis Report [FSER] Related to Certification of the AP600 Standard Design," on the basis that there were no missile hazards in the vicinity of the subject stairwells. External missile hazards were not considered in the staff's original evaluation of the AP600 stairwells. Following the events of September 11, 2001, the Federal Emergency Management Agency (FEMA) issued report FEMA 403, "World Trade Center Building Performance Study: Data Collection, Preliminary Observations and Recommendations," dated May 2002. Based on the performance of the gypsum stairwell enclosures in the World Trade Center following the aircraft impacts, Section 8.2.2.1 of the FEMA report recommends the use of impact-resistant enclosures around egress paths, such as stairwells. In light of the potential for external missile hazards, such as aircraft, the staff has re-considered its previous acceptance of gypsum stairwell enclosures in lieu of the concrete or masonry enclosure specified in the Branch Technical Position (BTP). Gypsum enclosures, while providing adequate fire resistance capability, are generally not considered impact-resistant to missiles. To address the concern about the performance of gypsum stairwell enclosures to potential missile hazards please perform a detailed evaluation of the vulnerability of each of the AP1000 stairwells located outside containment that serve as escape routes, access for firefighting, or access routes to areas containing equipment necessary for safe shutdown to external missile hazards. For those stairwells that are potentially vulnerable to external missiles provide a revision to the DCD to incorporate the original BTP guidance for the use of concrete or masonry enclosures. For those stairwells that are determined not to be vulnerable to external missiles provide a technical basis for that determination.

280.002

Section 9.5.1.2.1.1 states that the insulating and jacketing material for electrical cables meet the fire and flame test requirements of Institute for Electrical and Electronic Engineers Standard 1202 (IEEE 1202) or IEEE 383 excluding the option to use "flame source, oil or burlap." This

statement is not clear as the use of the “alternative” flame source allowed in Section 2.5.4.5 of IEEE 383 is not duplicated in IEEE 1202. During the review of the AP600 the staff approved the use of the 10-inch wide ribbon burner specified in IEEE 383 and IEEE 1202 as the only acceptable cable fire test procedure. Verify that the intent of the statement is solely to exclude the alternative flame source (i.e. oil soaked burlap) for use in the AP1000 as indicated in Item 96 of Table 9.5.1-1.

280.003

Section 9.5.1.2.1.1 states that the control of combustible materials is in accordance with National Fire Protection Association Standard 803 (NFPA 803), “Light Water Nuclear Power Plants.” NFPA 803 was withdrawn by the NFPA Standards Council in January 2001, subsequent to the issuance of NFPA 805, “Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants”; therefore, NFPA 803 is not an appropriate reference. The correct reference relating to the control of combustible materials for the AP1000 is Section 3.3 of NFPA 804, “Fire Protection for Advanced Light Water Reactor Electric Generating Plants,” 2001 Edition. NFPA 803 as well as superceded editions of other NFPA standards are cited in several other sections of the AP1000 DCD and WCAP-15871, “AP1000 Assessment Against NFPA 804.” Several of the editions cited in the DCD are different than the editions cited in the WCAP. SECY 93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advance Light Water Reactor (ALWR) Designs,” states that passive plant designs are to be reviewed using the latest industry standards endorsed by the Nuclear Regulatory Commission (NRC). Regulatory Guide (RG) 1.189, “Fire Protection for Operating Nuclear Power Plants,” provides a listing of the NFPA Codes and Standards endorsed by the NRC. Consistent with the criteria specified in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.34 (g), please revise the DCD and WCAP accordingly to reflect the applicable NFPA codes and standards that were in effect six months prior to the date of the AP1000 design certification application (March 28, 2002). This should include revisions to all citations to NFPA 803 and other superceded NFPA codes and standards.

280.004

Items 75 and 76 of Table 9.5.1-1 state that alternative or dedicated shutdown capability is not necessary. These statements are incorrect and conflict with Item 25 in the same table. As stated in NUREG-1512, Section 9.5.1.1.d, the staff concluded that the safety-related passive core cooling system (PXS) and passive containment cooling system (PCS) used to achieve and maintain safe shutdown following a fire in the AP600 are acceptable as an alternative/dedicated shutdown method for fire areas where the normal shutdown systems have not been protected in accordance with the guidance prescribed in the BTP. Please correct the discrepancy to be consistent with NUREG-1512 and Item 25 of Table 9.5.1-1 in the AP1000 DCD.

280.005

Item 198 of Table 9.5.1-1 states that safety-related battery rooms are separated from associated electrical rooms of the same division by one-hour fire rated barriers. NUREG-1512 states that safety-related battery rooms in the AP600 are separated from each other and other plant areas by three-hour fire rated barriers consistent with the guidance specified in Position C.7.g of the BTP. Provide a technical justification supported by mathematical fire

modeling for not providing three-hour fire barriers for the safety-related battery rooms in the AP1000, or revise the DCD to be consistent with the BTP and NUREG-1512.

280.006

Position C 5.c.(7) of the BTP states that the safe shutdown equipment and systems for each fire area be isolated from associated circuits such that hot shorts, open circuits, or shorts to ground will not prevent the operation of safe shutdown equipment and that a fire involving associated circuits will not prevent safe shutdown. Fires involving associated circuits may impact safe shutdown capability through loss of functions, flow diversions, blockage of flow paths, lost or misleading instrumentation, and loss of control. Consistent with this position, Section 9A.2.7.1 of the AP1000 DCD states that no postulated fire involving associated circuits will prevent safe shutdown; however, Section 9A.2.7.1 of the AP1000 DCD and Section 5.3.1.10 of WCAP-15871 states that only one worst case spurious actuation or signal results from a fire. These statements are not consistent with existing NRC guidance in (1) GL 86-10, “Implementation of Fire Protection Requirements,” Question 5.3.1, that states that for consideration of spurious actuations all possible functional failure states must be evaluated, (2) Question 5.3.8 that states that simultaneous high impedance faults for all associated circuits located in the fire area be considered, or (3) Section 9.5.1.5.c of NUREG-1512, which considered the potential for multiple spurious actuations resulting from a fire in the review of the AP600. Additional clarification on the staff’s position concerning circuit failures was provided to the nuclear industry in a March 11, 1997, letter to Mr. Ralph Beedle, Nuclear Energy Institute (NEI), from Mr. Samuel J. Collins, Director, Office of Nuclear Reactor Regulation. Section 9A.3.7.1 of the DCD addresses multiple spurious operations correctly in several systems but appears to be limited to high/low pressure interfaces. Please revise the AP1000 DCD and WCAP-15871 to be consistent with the staff’s positions concerning circuit failures and spurious actuations.

280.007

10 CFR 50.34(g) requires that applications include an evaluation of the facility against the guidelines of NUREG-0800, “Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants.” Section “I” of CMEB 9.5.1 of the SRP states that the purpose of the fire protection program is to provide assurance, through defense-in-depth, that a fire will not significantly increase the risk of radioactive releases to the environment in accordance with General Design Criterion (GDC) 3. Section 9.5.1.1.b of NUREG-1512 states the analysis demonstrates that the plant (AP600) will minimize radioactive releases to the environment. Contrary to these statements, Section 2.4 of WCAP-15871 indicates that an analysis of the potential effects of a fire on the release of contamination has not been included in the AP1000 fire protection analysis. Please resolve this discrepancy.

280.008

Section 57.4.1 of Chapter 57, “AP1000 Fire Risk Assessment” of the AP1000 Probabilistic Risk Assessment (PRA) (or the fire PRA) assumes the probability of a spurious signal impacting the automatic depressurization system (ADS) valves inside containment is an independent event. Section 5.3 of reference 10 (i.e., Circuit Analysis - Failure Mode and Likelihood Analysis issued by Sandia National Laboratory [SNL]) of the fire PRA states that the assumption that a given failure mode’s conditional probability value is actually independent remains a questionable

practice. Please provide a technical basis for the assumption that the probability of a spurious signal that has the potential to impact safe shutdown capability is independent.

280.009

Section 57.4.5.3.2 of the fire PRA only considers the potential effects of smoke from a fire on personnel performance, no assessment of the potential impacts on plant equipment has been provided. NUREG/CR-6597, "Results and Insights on the Impact of Smoke on Digital Instrumentation and Control," published in January 2001, concluded that smoke has the potential to be a significant environmental stressor that can result in adverse consequences. Please provide an analysis on the potential risk impacts of smoke on plant equipment.

280.010

Section 57.7.2 of the fire PRA evaluates the risk impact associated with the spurious actuation of ADS. No other spurious actuations have been addressed in the fire risk assessment. Fire-induced spurious actuations may impact safe shutdown capability through loss of system performance, flow diversions, blockage of flow paths, lost or misleading instrumentation, and loss of systems control. Provide a technical basis for excluding all other potential spurious actuations from the fire risk assessment, or provide an assessment of the risk impact for all potential spurious actuations that may prevent the operation or cause maloperation of systems needed to achieve and maintain safe shutdown for the AP1000.

280.011

Section 57.8 of the fire PRA states that the Containment (Fire Area 1000AF 01) core damage frequency (CDF) is an important plant contributor to the plant fire CDF. Table 57-9 indicates that approximately 41 percent of the total fire-induced CDF is assigned to the containment. Please provide a mathematical fire model (for each of the fire zones inside the Containment/Shield Building where redundant safe shutdown components required following a fire have not been separated by complete fire barriers) that supports the statements in the AP1000 DCD that a fire will be confined to the zone of origin such that the redundant components will remain free of fire damage. This includes the following fire zones: 1100 AF 111204, 1100 AF 11206, 1100 AF 11207, 1100 AF 11208, 1100 AF 11300A, 1100 AF 11300B, 1100 AF 11301, 1100 AF 11302, and 1100 AF 11500. Guidance on the application of fire modeling to nuclear power plant fire hazard analysis is provided in Appendix C of NFPA 805.

<u>Series 410 - Auxiliary Systems</u>

410.001

(DCD Sections 3.6.1 and 3.4.1) It is not clear to the staff what the design basis is for the protection against pipe breaks in non-seismically supported moderate energy lines for the AP1000. Item F of Section 3.6.1.1 of the DCD states that "[f]or systems not seismically analyzed for a safe shutdown earthquake, the safe shutdown earthquake is assumed to cause a pressure boundary failure." In Section 3.4.1.2.2, you identify in the second of six bullets that the (internal) flooding sources considered in the flooding analysis include, "moderate-energy piping (through wall cracks)." In the staff's AP600 FSER (NUREG-1512), Section 3.4.1.2, we listed the same 6 flood sources. However, in the FSER, the second bullet is identified as

"moderate-energy (*breaks and* through-wall cracks)," which implies that the staff believed full breaks were also considered in the flooding analysis for non-seismically supported moderate-energy lines.

The staff identified this concern in AP600 RAI 410.403F during the AP600 review. Your response indicated that you believed that this was a change in guidance and provided a response that indicated the plant design could withstand the flooding effects from a full pipe break (double-ended rupture) in any non-seismically supported moderate-energy pipe. However, it is not apparent that changes were made to the DCD to indicate that ruptures are assumed (and analyzed for environmental effects, i.e., flooding) in non-seismically supported moderate-energy piping.

(The staff does not consider this a change in guidance as we have always required that plants be capable of a safe shutdown following a full break (as a result of a seismic event) in any non-seismically supported moderate-energy line coupled with a worst-case single active failure. Section B.3.d of BTP 3.6.1, attached to SRP Section 3.6.1, Revision 1, is intended to identify the above as the staff's position with respect to non-seismically supported moderate-energy piping.)

Please verify that you have analyzed the AP1000 design for the flooding effects associated with full pipe breaks in non-seismically supported moderate energy piping systems, concurrent with any single active failure, and that the plant can still achieve and maintain a safe shutdown condition. A complete analysis is not necessary for those areas of the plant where the effects of such a break are obviously bounded by other piping systems in those areas. You should also revise the DCD to reflect this as the design basis.

NOTE: AP600 RAI 410.403F was issued by the staff on December 17, 1997 (NUDOCS Accession No. 9802040013). Westinghouse provided its response on January 9, 1998 (NUDOCS Accession No. 9801150055).

410.002

(DCD Section 3.6.1 and Appendix 3E) Appendix 3E of the DCD indicates that the hot water heating system (VYS) contains a limited amount of high energy piping in the auxiliary building (3-inch supply and return headers). You also state that "[t]here are no anchors or fittings on these lines in the nuclear island. Therefore, there are no postulated pipe breaks in these lines on the nuclear island." However, the VYS is identified as a Class E system (non-seismic) in Table 3.2-3 of the DCD. If the piping in this system is neither seismically analyzed nor seismically supported, an analysis must be performed to assess the effects of a postulated double-ended rupture of this piping (coupled with a single active failure) in areas with safe shutdown equipment and to assure that safe plant shutdown can still be achieved and maintained. A complete analysis is not necessary for areas where the effects would obviously be bounded by other pipe breaks in those areas. Please provide this analysis and revise the DCD as necessary.

410.003

Paragraph III.5.d of Section 10.3 of the SRP states that the main steam isolation valves, shut-off valves in connected connecting piping, turbine stop valves, and bypass valves should be able to close against maximum steam flow. Verify that these valves are capable of being closed against maximum steam flow.

410.004

(Section 10.4.1) Section 10.4.1.2.1 of the DCD states "[r]efer to Table 10.3.5-1 for permissible cooling water in-leakage and time of operation for maintaining the required condensate/feedwater quality." Where is this information found on Table 10.3.5-1?

410.005

(Section 5.2.5) AP1000 DCD, Section 5.2.5.3 indicates that the N_{13}/F_{18} radioactivity monitor can detect an 0.5 gallon-per-minute (gpm) leak when the plant is above 20 percent power. The detection sensitivity (0.5 gpm) is a function of the primary coolant radioactivity. Section 11.1 of the DCD discusses two source terms for the primary coolant: a conservative design-basis source term and a realistic source term. Please clarify which source term was assumed to determine the detection sensitivity of 0.5 gpm for the N_{13}/F_{18} radioactivity monitor. How can the assumption be verified with respect to the actual operating primary coolant radioactivity to assure the detection sensitivity of 0.5 gpm.

Position C.6 of RG 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," states that the response time of each leakage detection system should be adequate to detect a leak rate of 1 gpm, or its equivalent, in less than one hour. What is the response time for the N_{13}/F_{18} radioactivity monitor? Demonstrate the adequacy of this response time in meeting RG 1.45, Position C.6, and in supporting leak-before-break (LBB) for the AP1000.

410.006

(Section 5.2.5) TS 3.4.10, "RCS [reactor coolant system] Leakage Detection Instrumentation," specifies that (a) one containment sump level channel and (b) one containment atmosphere radioactivity monitor shall be operable for Modes 1, 2, 3, and 4. However, there are two NOTES allowing these two leakage detection instrumentation systems to not be required during certain conditions. The first note states that the containment atmosphere radioactivity monitor is only required to be Operable in Mode 1 with RTP [rated thermal power] > 20 percent. The second note states that containment sump level measurements cannot be used for leak detection if leakage is prevented from draining to the sump, such as by redirection to the in-containment refueling water storage tank (IRWST) by the containment shell gutter drains.

During Modes 1, 2, 3, and 4, if any one of these two conditions is satisfied and the instrument system is not used, will the RCS inventory balance (determined in accordance with surveillance requirement [SR] 3.4.8.1) or any other actions be required to compensate for the loss of diversity in TS 3.4.10? If not, justify the adequacy of this TS. When both conditions in these two notes are satisfied, and without compensatory actions, there will be no TS requirement for RCS leak detection such that LBB and small break loss-of-coolant accident (LOCA) may not be detected.

410.007

(DCD, Tier 2, Section 6.4, 9.4. through 9.4.3 and 9.4.6 through 9.4.11) The required aspects of a control room for nuclear power reactors are documented in 10 CFR Part 50, Appendix A,"General Design Criteria for Nuclear Power Plants." GDC 19, "Control Room," requires that a control room be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions.

Section 6.4.5.4 states that "[t]esting for main control room in-leakage during VES [main control room emergency habitability system] operation will be conducted once every 10 years. This testing will be conducted in accordance with ASTM [American Society for Testing and Materials] E741, 'Standard Test Method for Determining Leakage Rate by Tracer Dilution'." The staff is currently working with the industry to address control room habitability issues including air in-leakage testing. It is anticipated that the testing frequency will be on the order of 5 to 6 years. The staff expects that testing requirements for the AP1000 design will be consistent with the resolution of the control room habitability issues currently pursued by the industry and the staff. Therefore, the AP1000 design should include a commitment to resolving the in-leakage testing in accordance with the anticipated outcome of the joint effort between the NRC staff and industry. Please provide such a commitment and revise Section 6.4.8 to add the ASTM E741 standard.

In addition, consistent with the SRP, Westinghouse should commit to complying with the guidance contained in the latest versions of RG 1.52, "Design, Testing, and Maintenance for Post-Accident Engineered-Safety-Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," and RG 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

410.008

(DCD, Tier 2, Section 6.4 and Chapter 16 for TS 3.7.6) Chapters 6 and 16 state the following:

(1) Technical Specification Bases for SR 3.7.6.5: Verification of the initial air quality (in combination with other surveillances) ensures that breathable air is available for 11 main control room (MCR) occupants for at least 72 hours.

(2) Technical Specification Bases for SR 3.7.6.10: One VES air delivery flow path using the safety-related compressed air storage tanks, pressurizes the MCR envelope (MCRE) to at least a positive 1/8 inch water gauge pressure relative to the surrounding spaces at the required air addition flow rate of 65± 5 standard cubic feet-per-minute (scfm).

(3) Section 6.4.4 of the DCD: The VES maintains carbon dioxide (CO_2) concentration to less than 0.5 percent for up to 11 MCR occupants.

(4) Section 6.4.4 of the DCD: The VES nominally provides 65 scfm of ventilation air to the MCR from the compressed air storage tanks. Sixty scfm of ventilation flow is sufficient to pressurize the control room to at least positive 1/8-inch water gauge differential pressure with respect to

the surrounding areas in addition to limiting the CO_2 concentration below ½ percent by volume for a maximum occupancy of 11 persons and to maintain air quality within the guidelines of Table 1 and Appendix C, Table C-1, of Reference 1 ([American Society of Heating, Refrigerating, and Air Conditioning Engineers] ASHRAE Standard 62-1989).

Provide detailed justification as to why the AP600 MCRE design is equally applicable to the AP1000 design regarding compliance with the requirements of GDC 19 with respect to maintaining the safety-related radiation protection, toxic protection, and cooling functions. Since the AP1000 plant thermal rating is substantially higher than that of the AP600 design thermal rating, your detailed rationale should provide a discussion that includes, but is not limited to, the number of MCRE occupants, VES system capacity and capability, system redundancy to meet single failure criteria, safety-related system, structures, and components, and breathing air quality to meet U.S. Occupational Safety and Health Administration (OSHA) and ASHRAE Standards.

410.009

(DCD, Tier 1, Sections 2.2.5 and 2.7.1; Tier 2: Sections 6.4 and 9.4, and Chapter 16, TS 3.7.6 and TS 3.9.5): The NRC staff expects the AP1000 design to commit to compliance with the latest revisions of the applicable Codes and Standards for the following HVAC systems:

-radiologically controlled area ventilation system (VAS),
-non-radioactive ventilation system (VBS),
-containment recirculating cooling system (VCS),
-main control room emergency habitability system (VES),
-containment air filtration system (VFS),
-health physics and hot machine shop HVAC system (VHS),
-radwaste building HVAC system (VRS),
-turbine building ventilation system (VTS),
-annex/auxiliary buildings non-radioactive HVAC system (VXS), and
-diesel generator building heating and ventilation system (VZS)

Please review the applicable portions of the DCD descriptions and TSs to ensure proper references to the latest revisions to the applicable Codes and Standards and revise the DCD as necessary.

410.010

(DCD, Tier 2, Chapter 6, Section 6.4, and Chapter 9, Sections 9.4.1 through 9.4.3 and 9.4.6 through 9.4.11) The HVAC systems (VAS, VBS, VCS, VFS, VHS, VRS, VTS, VXS, AND VZS) were designed for a nominal 600 MW(e) plant. The same systems are credited for the 1000 MW(e) plant. Discuss the adequacy of these systems for the AP1000 design.

410.011

(DCD, Tier 2, Section 9.4.1) You stated in response to AP600 RAI 410.240 that Table 9.4-1 would be revised to summarize the plant areas served by nuclear filtration systems with their associated design/testing standards, filtration efficiency, design air flow rates, humidity control, charcoal adsorber thickness and maximum in-leakage flow. You also stated that the AP600

standard safety analysis report (SSAR) (DCD, Tier 2) would be revised to add Table 9.4-2 which would identify the minimum instrumentation and controls for nuclear filtration systems (as per RG 1.140, “Design, Testing, and Maintenance for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants”) based on American Society of Mechanical Engineers (ASME) Code N509 Table 4-2 criteria.

In the same AP600 RAI response, you also stated that the SSAR (AP600 DCD, Tier 2), Chapter 11 would be revised to reference Chapter 9, Table 9.4-1, to reflect gaseous radwaste management performance to state that “[i]n addition to the gaseous radwaste system release pathway, release of radioactive material to the environment occurs through the various building ventilation systems. These systems are described in Section 9.4 with a summary of system air flow rates and filter efficiencies provided in Table 9.4-1. The estimated annual release reported in Section 11.3.3 include contributions from the major building ventilation pathways.” These statements were to be inserted before Subsection 11.3.1 on page 11.3-1.

However, it appears that Table 9.4-1 was not revised to include the above information for the health physics area, radwaste building and radiation chemistry laboratory and that Table 9.4-2 was eliminated. Additionally, it appears that Section 11.3.1 was not revised to reflect the above information concerning the gaseous radwaste system. Please revise AP1000 DCD, Table 9.4-1, add Table 9.4-2, and revise Section 11.3.1 to reflect your AP600 RAI response and the above information or provide justification for their exclusion.

NOTE: AP600 RAI 410.240 was issued by the staff on May 23, 1994 (NUDOCS Accession No. 9406230015). Westinghouse submitted its response on July 22, 1994 (NUDOCS Accession No. 9407270187).

410.012

(DCD Tier 2, Section 9.4.1) The NRC staff’s RAI 410.371F, Item 2 for AP600 Tier 2 Material requested that Westinghouse “[s]tate in the text of the SSAR that the VES flow capacity conforms to: (1) the MCRE flow design “Table 1, and Appendix C Table C-1” of ASHRAE Standard 62-1989, “Ventilation for Acceptable Indoor Air Quality” and (2) 1993 ASHRAE Handbook, “Fundamentals SI Edition,” Chapter 23.2, “Ventilation and Indoor Air Quality,” since these references provide the appropriate guidelines for maintaining the carbon dioxide concentration limits below one-half percent by volume for a maximum occupancy of eleven persons inside the MCRE.” Westinghouse responded with the markups for AP600 SSAR (DCD Tier 2) Sections 6.4 and 9.4. However it appears that the AP1000 DCD Tier 2 Section 9.4.1.2.1.1 was not revised to include the proposed statement that “[t]he high air flow rates provided by the VBS system operation precludes the CO_2 concentration in the MCR exceeding the 0.05 % limit.” Please revise AP1000 DCD Tier 2 Section 9.4.1.2.1.1 (2nd sentence in the 3rd paragraph) accordingly or provide your rationale for its exclusion.

NOTE: AP600 RAI 410.371F was issued by the staff on December 8, 1997 (NUDOCS Accession No. 9712120340). Westinghouse submitted its response on December 29, 1997 (NUDOCS Accession No. 9801140150).

410.013

(DCD, Tier 2, Chapter 16, TS 3.7.6) The staff issued AP600 RAIs 410.371F, 410.378F, and 410.414F for the AP600 VES design and TS 3.7.6. As part of your responses to these RAIs, you committed to revise AP600 TS 3.7.6. It is not clear to the NRC staff that changes are reflected in the AP1000 DCD (TS 3.7.6). Please provide a discussion on the applicability of these AP600 RAI responses to AP1000 TS 3.7.6, and revise the AP1000 DCD/TSs accordingly.

NOTE: AP600 RAIs 410.371F and 410.378F were issued by the NRC staff on December 8, 1997 (NUDOCS Accession No. 9712120340). Westinghouse provided its response to these RAIs on December 29, 1997 (NUDOCS Accession No. 9801140150). AP600 RAI 410.414F was issued by the NRC staff on December 17, 1997 (NUDOCS Accession No. 98010503300). Westinghouse provided its response on January 15 and February 11, 1998 (NUDOCS Accession Nos. 9801230174 and 9802200316, respectively)

410.014

(DCD, Tier 2, Section 9.4) The staff issued AP600 RAIs 410.96 ,410.97, 410.99, 410.100, 410.101, 410.104, 410.241 and 410.243 addressing various AP600 ventilation systems. In response to these RAI questions, you provided specific clarifications as follows:

A. AP600 RAI 410.96: Westinghouse indicated that safety-related MCR isolation dampers in VBS conform to the Positions C.1 and C.2 of RG 1.29 as discussed in SSAR Appendix 1A,

B. AP600 RAI 410.97: Westinghouse indicated that safety-related MCR isolation dampers and the VBS design conform to GDC 4, “Environmental and Missile Design Basis,”

C. AP600 RAI 410.99: Westinghouse indicated that the fuel handling area HVAC subsystem of VAS conforms to Positions C.1 through C.4 of RG 1.29 and VFS conforms to Positions C.1.a through C.1.d, and C.2.a through C.2.f, of RG 1.140 and Position C.4 of RG 1.13,

D. AP600 RAI 410.100: Westinghouse indicated that the VFS containment penetrations that provide containment isolation conform to single failure criteria based on SRP Section 9.4.5 guidelines,

E. AP600 RAI 410.101: Westinghouse indicated that VRS conforms to Positions C.1 and C.2 of RG 1.140,

F. AP600 RAI 410.104: Westinghouse indicated that VHS conforms to Positions C.1 and C.2 of RG 1.140,

G. AP600 RAI 410.241: Westinghouse provided tabulated flow data for VAS, VBS, VHS, and VRS to demonstrate that these systems maintain their served areas at certain positive or negative pressures with respect to surrounding spaces and/or outdoor atmosphere, and

H. AP600 RAI 410.243: Westinghouse provided the rationale for the installation of filtration air supply fans upstream of filtration units for VBS in order to meet the requirements of Section 4.7.2, “Habitability Systems,” of ASME N509-1989.

It is not clear to the NRC staff that the above clarifications are appropriately reflected in the AP1000 DCD. Please clarify each issue with regard to the applicability to the AP1000, DCD Tier 2 Section 9.4 design, and revise the AP1000 DCD, Tier 2 Section 9.4 accordingly.

AP600 RAIs 410.96, 410.97, 410.99, 410.100, 410.101, and 410.104 were issued by the NRC staff on January 26, 1993 (NUDOCS Accession No. 9303050146). AP600 RAIs 410.241 and 410.243 were issued by the NRC staff on May 23, 1994 (NUDOCS Accession No. 9406230033). Westinghouse provided its response to RAIs 410.96 and 410.97 on March 30, 1993 (NUDOCS Accession No. 9304050085); to RAI 410.100 on April 29, 1993 (NUDOCS Accession No. 9305050162); to RAI 410.99, 410.101, and 410.104 on May 14, 1993 (NUDOCS Accession No. 9305190237); to 410.241 on July 15, 1994 (NUDOCS Accession No. 9407250275); and to 410.243 on July 29, 1994 (NUDOCS Accession No. 9408040228).

410.015

(DCD Tier 2, Sections 6.4 and 9.4) During the AP600 design certification review, the NRC staff requested (in AP600 RAI 410.415F) that Westinghouse provide clarification regarding the location (i.e., located in the MCR, local, or both) of the system instrumentation (such as pressure indications and high differential pressure alarms for the system filters and unit coolers, airflow indication and alarms to monitor operation of the supply and exhaust fans, etc.), for the HVAC systems (VBS, VXS, VAS, VCS, VFS, VRS, VTS, VHS, and VZS).

You stated in your response to the above AP600 RAI that: “The A600 has a plant-wide network that provides pre-processed plant data to those locations where the information is required. Because of the rapid changes that are taking place in the digital computer and graphic display technology employed in a modern human system interface, design certification of the AP600 focuses upon the process used to design and implement human system interfaces for the AP600 rather than on the details of the implementation. As a result, SSAR Chapter 18 describes the processes used to provide human factors engineering in the design of the AP600. The specifics of display and alarm will be developed as part of the human factors implementation process. In general, variables discussed in this question are expected to be available both in the control room and at remote “data port” locations throughout the plant with the use of a portable data display device.”

Please clarify in the AP1000 DCD system descriptions (including VES, VBS, VXS, VAS, VCS, VFS, VRS, VTS, VHS, and VZS) where the instrumentation information is provided; is the information provided locally, or in the main control room (MCR), or is it provided in both places? Provide the rationale for its exclusion.

NOTE: AP600 RAI 410.415F was issued by the staff on December 18, 1997 (NUDOCS Accession No. 9802030132). Westinghouse submitted its responses on January 1 and January 27, 1997 (NUDOCS Accession Nos. 9801130177 and 9802050165).

410.016

The second paragraph of Section 9.1.3.4.3 “Abnormal Conditions,” states that, in the unlikely event of an extended loss of normal spent fuel pool (SFP) cooling, a water level is maintained above the spent fuel assemblies for at least 7 days and that the amount of makeup required to

provide the 7 day capability depends on the decay heat level of the fuel in the SFP and is provided “when the calculated decay heat level in the spent fuel pool is less than 2.3 MWt [megawatts thermal], no make up is needed to achieve spent fuel pool cooling for at least 7 days.”

Please describe the mechanism that the operator will use to determine the thermal power level in the SFP. For example, if the calculated decay heat level in the SFP is 2.4 MWt, how would the operator know? Are means provided for the operator to read this value in the control room? Will the operator be required to perform the calculation?

410.017

Section 9.1.3.4.3.3, “Loss of Offsite Power,” states that the SFP cooling system pumps can be manually loaded on the respective onsite standby diesel generator in the event of a loss of offsite power. In the AP600 design this is an automatic operation (which the staff found acceptable). Under the circumstances of a loss of offsite power, it is reasonable to assume that the operator will likely be quite busy. Please provide the rationale and technical justification for making this a manual operation.

410.018

Section 9.1.3.4.3.4, “Station Blackout,” states that water vapor that evaporates from the surface of the SFP is vented to the outside environment through an engineered relief panel. This vent path maintains the fuel handling area at near atmospheric pressure conditions. Activity releases due to pool boiling are analyzed. Please discuss the method of analyzing the releases and provide details describing exactly how the activity is captured for analysis.

410.019

Section 9.1.3.5, “Safety Evaluation,” states that the SFP is designed such that a water level is maintained above the spent fuel assemblies for at least 3 days following a loss of the SFP cooling system, using only safety-related makeup. In the AP600 design, under similar circumstances, water is maintained above the spent fuel assemblies for 7 days (which the NRC staff found acceptable). Please discuss the rationale for maintaining water above the spent fuel assemblies for only 3 days following a loss of the SFP cooling system, and provide the technical justification describing the basis for acceptability of the 3-day duration.

410.020

Section 9.1.3.5, “Safety Evaluation,” states that safety-related makeup water can be supplied to the SFP from the fuel transfer canal, cask washdown pit, and passive containment cooling water storage tank (PCCWST). Since the passive containment cooling water storage tank is non-safety-related, please explain the circumstances that permit safety-related makeup water to be provided from a non-safety-related source.

410.021

Provide appropriate verifications for the concerns identified below for the subject AP1000 HVAC systems and/or identify where these verification discussions are provided in the AP1000 DCD:

A. In response to the AP600 RAI 410.241, a summary of ventilation flows and corresponding ambient pressure data were provided for the AP600 design for the VAS, VBS, VHS and VRS. Are these data directly applicable to the AP1000. Please provide such data for these systems for the AP1000.

B. In response to the AP600 RAI 410.245.c, VZS was defined as a defense-in-depth system and conformed to the staff's identified criteria for non-safety-related systems requiring regulatory controls, i.e., RTNSS systems. Verify that this information is equally applicable to the AP1000 VZS design (is this system similarly classified a RTNSS system as it was in the AP600?).

C. AP600 DCD Tier 2 Section 9.4.7.1.2 states that VFS provides filtration of exhaust air from the fuel handling area, auxiliary or annex building to maintain these areas at a "slight negative pressure" with respect to the adjacent areas. In response to the AP600 RAI 410.345F, "slight negative pressure" is defined as a "nominal set point value of (negative) 0.15-inch water gauge (WG) of the differential pressure)." Verify that this information is equally applicable to AP 1000 VFS design.

NOTE: AP600 RAI 410.241 410.245 were issued by the NRC staff on May 23, 1994 (NUDOCS Accession No. 9406230033). AP600 RAI 410.345 was issued by the NRC staff on December 9, 1997 (NUDOCS Accession No. 9801260106). Westinghouse provided its response to RAI 410.241 on July 15, 1994 (NUDOCS Accession No. 9407250275); to RAI 410.245 on May 7, 1997 (NUDOCS Accession No. 9705280194); and to RAI 410.345 on December 12, 1997 (NUDOCS Accession No. 9712230390).

<u>Series 460 - Effluent Treatment</u>

460.001

(Sections 11.1, 11.2, and 11.3) The assumption of 0.25 percent fuel defect described in Section 11.1.1.1 for the AP1000 design-basis source term deviates from the fuel defect assumption of 1.0 percent described in SRP Sections 11.2 and 11.3 for the liquid waste and gaseous waste management systems. Please address the following:

A. Justify this deviation, and

B. In the AP1000 DCD, the compliance of liquid and gaseous effluent concentration in unrestricted area with 10 CFR Part 20 limits is not based on a 1 percent fuel failure on the annual average. Table 11.2-9 and 11.3-4 of the DCD should be re-evaluated based on a 1 percent fuel failure and annual average effluent concentrations of radionuclides in unrestricted areas.

460.002

(Section 11.1) Demonstrate that the AP1000 design meets the following criteria listed in SRP Section 11.1, Paragraph II, “Acceptance Criteria.” Please identify where each of the following criteria are addressed in the DCD. If any criteria is not addressed, please provide that information and revise the DCD accordingly. Providing a table that identifies the applicable criteria and DCD section would be adequate.

A. The parameters used to calculate primary and secondary coolant concentrations are consistent with those given in NUREG-0017, “Calculation of Releases of Radioactive Materials in Liquid and Gaseous Effluents from Pressurized Water Reactors.”

B. All normal and potential sources of radioactive effluent delineated in Subsection I of SRP Section 11.1 are considered.

C. For each source of liquid and gaseous waste considered in subsection I.1 of SRP 11.1, the volumes and concentrations of radioactive material given for normal operation and anticipated operational occurrences are consistent with those given in NUREG-0017.

D. Decontamination factors for in-plant control measures used to reduce gaseous effluent releases to the environment, such as iodine removal systems and high-efficiency particulate air (HEPA) filters for building ventilation exhaust systems and containment internal cleanup systems are consistent with those given in RG 1.140. The building mixing efficiency for containment internal cleanup is consistent with NUREG-0017.

E. Decontamination factors for in-plant control measures used to reduce liquid effluent releases to the environment, such as filters, demineralizers, and evaporators are consistent with those given in NUREG-0017.

F. Radwaste augments used in the calculation of effluent releases to the environment are consistent with the findings of a cost-benefit analysis, and are performed using the guidance of RG 1.110, “Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Plants.” The provisions that require a cost-benefit analysis are stated in Section II.D of Appendix I to 10 CFR Part 50.

G. Effluent concentration limits at the boundary of the unrestricted area do not exceed the values specified in Table 2 of Appendix B to 10 CFR Part 20.

H. The source terms result in meeting the design objectives for doses in an unrestricted area as set forth in Appendix I to 10 CFR Part 50.

I. For evaluating the source terms, the applicant should provide the relevant information in the DCD as required by 10 CFR Part 50, Section 50.34a. This technical information should include all the basic data listed in Appendix B to RG 1.112 required in calculating the releases of radioactive material in liquid and gaseous effluents. An acceptable method for satisfying the criteria given in items 1 through 6 consists of using the Gaseous and Liquid Effluent (GALE) Computer Code and the source term parameters given in NUREG-0017.

J. If the calculational technique or any source term parameter differs from that given in NUREG-0017, please describe these differences in detail and the bases for the methods and parameters used.

460.003

(Sections 11.2, 11.3, and 11.4) Please provide additional information regarding RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants." Specifically, address the following:

A. According to DCD Appendix 1A relating to RG 1.143, the radwaste building of the AP1000 is designed to Uniform Building Code (UBC)-1991. This is a deviation from Criterion C6.2.1 of RG 1.143, Revision 2, November 2001, which refers to UBC-1997, ASCE (American Society of Civil Engineers) 7-95. However, Section 3.7.2 of the AP1000 DCD refers the building code to UBC-1997. Please clarify the inconsistency and update the DCD accordingly.

B. Section 5 of RG 1.143, Revision 2 defines the classification (RW-IIa, RW II-b, and RW II-c) of radwaste systems for design purposes. In Appendix 1A of AP1000 DCD relating to the conformance with RG 1.143, Revision 2, Westinghouse indicated that systems containing enough activity to possibly be classified as RW-IIa are located in the auxiliary building (which is Seismic Category I). AP1000 conforms with Positions C5.2 and C5.4 with respect to Class RW-IIb. Please identify Class RW-IIb systems, components, and structures in the AP1000 design.

460.004

(Section 11.2) AP1000 DCD, Section 11.2.1.2.1, "Capacity," states that the liquid waste system provides adequate capacity to meet the anticipated processing requirements of the plants. The tables being referenced in Section 11.2.1.2.1 do not have sufficient explanation regarding the capacity of the liquid waste system. The projected flows of various liquid waste systems under normal conditions are identified in DCD Table 11.2-1. Please provide additional information that demonstrates the adequacy of the liquid waste system to handle the normal input specified in Table 11.2-1.

DCD Table 11.2-4 contains information on the surge capacity of individual tanks. Where is the information about the surge input rates? Demonstrate the adequacy of the capacity to handle the surge input.

Revise DCD Section 11.2.1.2.1 to specifically explain the capacity of the liquid waste management system.

460.005

(Section 11.3) Please provide additional information addressing the following for the gaseous radwaste management systems:

A. demonstration of compliance with BTP ETSB 11.5, with a short-term (0-2 hours) release duration,

B. a discussion of compliance with GDC 3, “Fire Protection,” as it relates to providing protection to gaseous waste handling and treatment systems from the effects of an explosive mixture of hydrogen and oxygen. The discussion should include the provisions incorporated in the AP1000 design to control releases due to hydrogen explosions in the gaseous waste management team. Additionally, it should include the type, number and locations of gas analyzers provided in the design of the gaseous waste management system,

C. a discussion of compliance with GDC 60, “Control of Releases of Radioactive Materials to the Environment,” as it relates to control of releases of radioactive materials to the environment. The discussion should refer to RG 1.140 and should be consistent with Acceptance Criterion II.6.a of Section 11.3 of the SRP, and

D. a discussion of compliance with GDC 61, “Fuel Storage and Handling and Radioactivity Control,” as it relates to radioactivity control in gaseous waste management systems and ventilation systems associated with fuel storage and handling areas.

460.006

(Section 11.4) Discuss the specific design features provided in the system design to comply with GDCs 60, 63, “Monitoring Fuel and Waste Storage,” and 64, “Monitoring Radioactivity Releases,” as they relate to (1) Acceptance Criteria in SRP Section 11.4, Paragraph II.C, regarding control of release of radioactive materials to the environment from the plant areas where the solid radwastes are processed, and (2) Acceptance Criteria in SRP Section 11.4, Paragraph II.D, regarding the radioactive waste system being designed for monitoring radiation levels.

460.007

(Section 11.4) Staff guidance for the radioactive waste storage capacity that is sufficient to allow time for short-lived radionuclides to decay prior to shipping is discussed in SRP Section 11.4, Paragraphs II.6 and III.4, and BTP ETSB II-3, Position B.III. Please demonstrate that the AP1000 design has such sufficient storage capacity. Clarify which generation rate (expected or maximum) is used for this demonstration. Identify the in-plant storage space (spent resin tanks, spent filter tubes,...etc.) associated with different kind of wastes (wet wastes, dry wastes,...etc.), shipment capacity, and duration for in-plant storage.

460.008

(Section 11.4) In Generic Letter (GL) 81-38, “Storage of Low-Level Radioactive Wastes at Power Reactor Sites,” the staff provided guidance to licensees on the addition of on-site storage facilities for low-level radioactive wastes generated on-site. The staff recognizes that the need for additional on-site storage facilities is a site-specific issue. However, this is not identified in Section 11.4.6 of the AP1000 DCD as COL action items. Please include it in Section 11.4.6 as a COL action item.

460.009

Section 11.4 of the AP1000 DCD states that solid waste management system does not handle large, radioactive waste materials such as core components or radioactive process wastes from the plant's secondary cycle. Where are these large, radioactive waste materials handled?

460.010

SRP Section 11.5, Paragraph II.2.c of the acceptance criteria states that provisions should be made to purge and drain sample streams back to the system of origin or to an appropriate waste treatment system. Please demonstrate that the AP1000 design meets this criterion.

460.011

SRP Section 11.5, Paragraph II.3 of the acceptance criteria states that provisions should be made for administrative and procedural control, for necessary auxiliary or ancillary equipment, for special features for the instrumented radiological monitoring sampling, and for analysis of process and effluent streams. Please demonstrate that the AP1000 design meets this criterion.

Series 480 - Containment Systems

480.001

(DCD Section 6.2.4, "Containment Hydrogen Control System") In the redline-strikeout version of Section 6.2.4.2.2, "Hydrogen Recombination Subsystem," the following paragraph has been struck out:

> A third PAR [passive autocatalytic recombiner] is located at one of the vent paths from the IRWST and is utilized to limit the accumulation of hydrogen within the IRWST during normal and post-accident operation.

NRC Information Notice (IN) 2002-15, "Hydrogen Combustion Events in Foreign BWR Piping," dated April 12, 2002 (ADAMS Accession No. ML020980466), describes two events in which hydrogen gas accumulated in pipes during normal plant operation and then exploded, destroying several feet of pipe. Although the events occurred at boiling water reactors (BWRs), the guidance in the IN was addressed to all operating light-water reactors. The following paragraph is taken from the IN:

> These events show the importance of preventing combustible gas mixtures from accumulating in piping. In both of the above described events, hydrogen and oxygen gases apparently accumulated to a combustible level which then catastrophically failed these piping systems. Proper venting or other considerations to prevent accumulation of combustible gases in piping high points might alleviate conditions leading to hydrogen combustion.

Furthermore, GDC 3, “Fire Protection,” states, in part:

> Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

Finally, SRP Section 11.3, “Gaseous Waste Management Systems,” provides guidance concerning protection of gaseous waste handling and treatment systems from the effects of an explosive mixture of hydrogen and oxygen. Although the IRWST is not part of a gaseous waste handling and treatment system per se, there is a potential for an accumulation of hydrogen gas and the development of a combustible or explosive mixture.

Considering that the PAR in the IRWST in the AP600 design was provided to limit the accumulation of hydrogen in the IRWST during normal operation, justify the decision to not include that PAR in the AP1000 design, given the requirements of GDC-3 and the issues raised in IN 2002-15.

480.002

(DCD Section 6.2.3, Containment Isolation System”) This section makes no mention of RG 1.141, “Containment Isolation Provisions For Fluid Systems,” dated April 1978, which endorses, with exceptions, ANSI [American National Standard Institute] N271-1976/ANS-56.2, “Containment Isolation Provisions For Fluid Systems,” dated June 28, 1976.

Provide a discussion of the AP1000 design’s conformance to the provisions of this RG.

480.003

DCD Section 6.2.1.1.3, “Design Evaluation,” states that the passive internal heat sink data used in the WGOTHIC analyses is presented in DCD Reference 20, Section 4. Section 4 of DCD Reference 20 describes the AP600. Section 13 of DCD Reference 20 describes the AP1000. Please revise the DCD to identify the correct section for the description of the AP1000 WGOTHIC model.

480.004

DCD Section 6.2.1.1.3, “Design Evaluation,” states that heat conduction from the wet to dry section is considered in the analysis, referring to Section 7 of DCD Reference 20 (the 2-dimensional [2-D] conduction effect described in Section 7.4). While no credit for pressure reduction is taken for the source term evaluation, consistent with the staff’s approval of WGOTHIC in NUREG-1512, the inclusion of this effect prevents the staff from making a finding regarding compliance with GDC 38, “Containment Heat Removal.” Please provide an analysis for the limiting LOCA case without the 2-D conduction correction to the PCS water flowrate (area coverage). Please provide the revised evaporation-limited PCS flow and a comparison plot to the PCS flowrate with the 2-D conduction effect (similar to Figure 13-93 in DCD Reference 20). Please provide a plot of the pressure response for at least the first 24 hours into the event.

480.005

Section 6.2.1.2.3, “Design Evaluation,” indicates that DCD Reference 26 contains the details for the subcompartment pipe break analyses. DCD Reference 26 does not have a document identifier or a document date. By its name, it implies that no changes to the subcompartment models were made to evaluate the AP1000. Please provide a copy of the reference for staff review.

480.006

Please identify the break size (and location) for DCD Table 6.2.1.3-4. Is this the 1.0 square foot (ft^2) break identified in DCD Section 6.2.1.2.3.2, “Pipe Breaks”? Please revise the DCD to include this information in the table. DCD Table 6.2.1.3-4 also is not referenced in DCD Table 6.2.1.2-1. Please revise the DCD appropriately.

480.007

Please provide the data presented in DCD Tables 6.2.1.3-9 and 6.2.1.4-2 in an electronic format for use by the staff to perform independent confirmatory analyses. Also, please provide the data presented in Figure 7-2, “Minimum Delivered PCS Water Flow Rate,” of WCAP-15846, “WGOTHIC Application to AP1000,” for the AP1000 in an electronic format for use by the staff to perform independent confirmatory analyses.

480.008

In DCD Section 6.2.1.3.2.1, “Mass and Energy,” and in DCD Section 6.2.1.4.1.1, “Plant Power Level,” you state that the power level used for the design-basis accident (DBA) analyses is taken to be 101 percent (of full power to account for a 1 percent calorimetric error). This is not consistent with the guidance provided in SRP Sections 6.2.1.3, “Mass and Energy Release Analysis for Postulated Loss-of-Coolant,” and 6.2.1.4, “Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures,” which indicate that the power level should be taken as 102 percent of full power. This is also inconsistent with the approved methods used by Westinghouse. For example in DCD Reference 3, Section II, it is stated that the core stored energy is based on a conservative value of 102 percent of the engineering safeguards design rated power level. This is also inconsistent with the analyses described in DCD Section 6.2.1.5, “Minimum Containment Pressure Analysis for Performance Capability Studies of Emergency Core Cooling System,” which references DCD Section 15.6.5 wherein the 2 percent calorimetric error is also used (DCD Table 15.6.5.-4). The staff is aware that the July 31, 2000, revision to Appendix K to 10 CFR Part 50 allows for the justification of a power level lower than 102 percent for demonstrating compliance with 10 CFR 50.46. The staff is also aware that there is a potential inconsistency with the revised rule and other regulatory guidance. Please provide a new analyses for the limiting pipe breaks to demonstrate that the calculated containment peak pressure and temperature and, if applicable sub-compartment loads, remain acceptable at 102 percent of full power. As an alternative you may provide a justification for review by the staff for the use of the 1 percent calorimetric error for the containment DBA calculations as well as justification for departing from the current SRP guidance and for departing from your approved mass and energy release methodology.

480.009

There are several discrepancies (as cited below) in the DCD regarding containment purge isolation valve closure time.

- DCD Tier 2, Table 14.3-7 (Sheet 1 of 3), “Radiological Analysis,” indicates that the maximum closure time for remotely-operated containment purge valves is ≤ 5 seconds.

- DCD Tier 2, Table 6.2.3-1 (Sheet 3 of 4), “Containment Mechanical Penetrations and Isolation Valves,” indicates that the closure time for the containment purge valves (VFS-PL-V003, -V004, -V009, and -V010) is 20 seconds.

- DCD Tier 1, Table 2.2.1-3, “Inspections, Tests, Analyses, and Acceptance Criteria,” Item 7 indicates that the containment purge isolation valves (VFS-PL-V003, -V004, -V009, and -V010) close within 10 seconds.

- DCD Tier 2, Section 6.2.1.5, “Minimum Containment Pressure Analysis for Performance Capability Studies of Emergency Core Cooling System” (PWR), sub-section 6.2.1.5., indicates that, to help minimize containment pressure, the containment purge supply and exhaust lines were modeled to close 12 seconds after the 8 psig closure setpoint was reached (apparently based on a 10 sec closure time, increased by a factor of 1.2).

Please clarify this issue and correct the apparent discrepancies in the DCD.

<u>Series 640 - Inspections, Tests, Analysis, and Acceptance Criteria (ITAAC)</u>

640.001

Please reconcile the apparent inconsistency between (1) the minimum value of 755,100 gallons stated in DCD Table 6.2.2-1, for PCCWST useable capacity for the PCS (gal) and (2) the acceptance criteria of “greater than 755,000" gallons stated in DCD ITAAC Table 2.2.2-3, Item 7.(f)ii.

HISTORY OF PREVIOUSLY-ISSUED
REQUESTS FOR ADDITIONAL INFORMATION

Letter No.	Date issued	ADAMS Accession No.	RAI Nos.	Date of response	ADAMS Accession No.
1	6/26/2002	ML021780568	440.001 - 440.008	7/24/2002	ML022110430
2	8/16/2002	ML022280379	720.001	9/10/2002	
3	8/27/2002	ML022390103	420.001 - 420.046, 435.001 - 435.015		
4	9/3/2002	ML022460356	620.001 - 620.043		
5	9/4/2002	ML022470255	210.001 - 210.057		
6	9/5/2002 Reissued 9/18/2002	ML022480440, Reissued RAI ML022610042	440.009 - 440.148, 720.002 - 720.026	9/12/2002 (440.009)	ML022600097
7	9/19/2002	ML022620026	260.001 - 260.003, 261.001 - 261.010, 471.001, 471.010, 472.001, 472.003		
8	9/19/2002	ML022620319	220.001 - 220.019, 230.001 - 230.019, 240.001 - 240.004, 241.001 - 241.003		

HISTORY OF PREVIOUSLY-ISSUED
REQUESTS FOR ADDITIONAL INFORMATION

9	9/24/2002	ML022620079	250.001 - 250.003, 251.001 - 251.029, 252.001 - 252.009, 281.001- 281.003		
10	9/25/2002	ML022620614	210.058 - 210.070, 261.011 - 261.013, 720.027 - 720.040		
11	9/27/02	ML022670315	440.149 - 440.182, 451.001 - 451.007, 470.001 - 470.013, 720.041 - 720.097		

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