September 25, 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. 10 - 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 topics covered in these requests for additional information (RAIs) include the areas of mechanical engineering, initial test program, and reliability and risk assessment. These RAIs were sent to you via electronic mail on September 6, September 17, and September 20, 2002. RAI No. 210.065 was discussed with your staff during a meeting held from September 9 through September 11, 2002, at Westinghouse Energy Center. 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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Request for Additional Information (RAI)
AP1000 Standard Design Certification
Series 210 - Mechanical Engineering
Series 261 - Initial Test Program
Series 720 - Reliability and Risk Assessment

Series 210 - Mechanical Engineering

Reference: AP1000 Design Control Document (DCD), Highlight/Strikeout Version, APP-GW-GL-701 Rev.0, dated January 2002.

Review Subject: Reactor Vessel Control Rod Drive System

210.058

Reference, Volume 6, Section 3.9.4.1.1, Page 3.9-67, third paragraph (¶):

The four major subassemblies of the control rod drive mechanism (CRDM) are identified, and their functions are discussed, but the arrangement of the interfacing subassemblies shown in Figure 3.9-4 is not clearly defined.

Please provide clarification of Figure 3.9-4 including identification of the four separate subassemblies discussed, and specific identification of the portion of the CRDM which is included in the American Society of Mechanical Engineers (ASME) Code Section III, Class 1 reactor coolant pressure boundary.

210.059

Reference, Volume 6, Section 3.9.4.2.1, Page 3.9-72, second bullet:

Please indicate how the 166.755 inch travel dimension is determined, and discuss the implication of the three significant figures used for the fractional part of the dimension specifying maximum travel.

210.60

Reference, Volume 6, Section 3.9.4.2.1, Page 3.9-72, third bullet:

The 400-pound maximum load capability does not agree with the CRDM load capacity specified on Page 3.9-68, second paragraph. Please clarify.

Also, please identify the maximum total weight of the rod cluster control and drive rod assembly which is actually raised or lowered by each CRDM.

210.061

Reference, Volume 6, Section 3.9.4.2.3, Page 3.9-73:

(Editorial): The first sentence in this section appears to be missing some text, or may be incorrectly worded. Please provide clarification.

210.062

Reference, Volume 6, Section 3.9.4.2.3, Page 3.9-73, fourth ¶:

The discussion states that the design of CRDM non-pressure boundary components are based on material specification mechanical property requirements. However, material property specifications typically specify minimum required yield stress and minimum required tensile ultimate stress, but do not include allowable stress criteria for design purposes.

Please provide additional discussion identifying the allowable stress criteria used for the design of CRDM non-pressure boundary components, including design considerations given toward the establishment of a 60-year design life for the moving components of the CRDM.

210.063

Reference, Volume 6, Section 3.9.4.3, Page 3.9-75, third ¶:

The discussion of the analysis to assure functional capability of the CRDM does not clearly define the acceptance criteria.

Please provide additional details of the analysis, including definition of the ASME Code Section III Service Level stress limits used to define the allowable bending moments and the allowable deformations of the CRDM. Also identify the limiting, worst-case postulated pipe rupture used for analysis of the functional capability of the CRDM.

Additionally, please provide an explanation justifying the use of ASME Code Section III stress criteria (normally used for establishing structural integrity) for the purpose of assuring component functional capability.

210.064

Reference, Volume 6, Section 3.9.4.4, Page 3.9-75, fifth ¶:

Reference is made to functional test programs that have been conducted to confirm the operational capability of the CRDM.

Please provide a summary of results of these tests with comparisons of test results to design acceptance criteria, including criteria for demonstration of CRDM operational capability following exposure to the combined effects of a loss-of-coolant (LOCA) accident and a safe shutdown earthquake.

210.065

Hydrodynamic loads could be significant for piping systems with fast opening and closing valves. The reactor vessel head vent system (RVHVS) piping has been identified as a system susceptible to such large hydraulic transient loadings. Experience from operating nuclear power plants indicates that thermal hydraulic loads could be very severe for certain operating modes if the piping layout does not take this aspect into careful consideration. Please provide assurance that the AP1000 plant specific RVHVS piping configuration can be designed to withstand the transient thrust forces caused by the valve opening without performing the thermal hydraulic and structural dynamic analyses at this time. To demonstrate the conceptual design and facilitate the staff's review, it is suggested that a sketch be provided along with the response.

NOTE: This question was discussed with Westinghouse representatives in a meeting at Westinghouse Energy Center held from Monday, September 9, 2002, through Wednesday, September 11, 2002.

210.066

AP1000 DCD, Revision 0, Volume 3, "Tier 2 Information" did not address New Generic Issue 89, "Stiff Pipe Clamps." Please provide information on whether the use of stiff clamps are allowed or prohibited on the design of the AP1000 piping systems. If the use of stiff pipe clamps are permitted, please define the conditions under which they may be used and describe how their impacts on stiffness and discontinuities are addressed on the design of thick-wall and thin-wall piping systems.

210.067

NRC Bulletin 87-01, "Thinning of Pipe Walls in Nuclear Power Plants," and Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning," requested nuclear power plant licensees to implement a program to ensure that erosion/corrosion does not lead to degradation of single-phase and two-phase high-energy carbon steel systems. The main feedwater system was highlighted as an example of a pipe line that is susceptible to this type of degradation mechanism; also referred to as flow-accelerated corrosion (FAC). This type of problem should be prevented, to the extent practical, at the design stage to minimize the potential impacts. Please provide additional information discussing design measures for the AP1000 plant to minimize the effects of FAC on the feedwater line inside and outside the containment, including selection of material that is less susceptible to FAC, and component type and layout considerations.

210.068

Reference: WCAP-15800, "Operational Assessment for AP1000," Revision 0, dated April 2002, Pages 2-2 and 2-3.

The resolution comment provided for Bulletin 81-01, "Failure of Mechanical Snubbers," and for Bulletin 81-01, Revision 1, "Surveillance of Mechanical Snubbers," refers to AP1000 DCD Section 3.9.6. However, DCD Section 3.9.6 addresses only inservice testing of pumps and valves, and does not include any information on mechanical snubbers. It is acknowledged that

Bulletin 81-01 dealt with examinations of snubbers installed in operating plants and this aspect of the bulletin is not applicable to the AP1000 design certification. However, the surveillance and qualification testing implications of these bulletins should be addressed during the design certification process. DCD Section 3.9.6 does not provide this information. Please provide additional discussion or references, including commitments to current documents which address the surveillance and testing of mechanical snubbers used in the AP1000 design.

210.069

Reference: WCAP-15800, Page 3-7

The resolution comment provided for GL 80-109, "Guidelines for SEP and Soil Structure Interaction Reviews," states that the AP1000 has been designed for a range of soil conditions, and refers to DCD Section 3.7.2. Please provide a revised resolution comment to be consistent with the limitations placed on the AP1000 design for certification for hard rock sites only. (As stated in your letter dated February 13, 2002, ADAMS Accession No. ML020640065. This issue was also discussed in a telephone conference on August 22, 2002, between Westinghouse personnel and the NRC staff, in which it was concluded that the AP1000 DCD Section 3.7 would be revised to remove all discussions of seismic analysis using soil-structure interaction methodology.)

210.070

Reference AP1000 DCD, Revision 0, dated January 2002, Section 1.9.4.2.3, "New Generic Issues."

The "AP1000 Response" to Issue 113, "Dynamic Qualification of Large-Bore Hydraulic Snubbers," provided on DCD page 1.9-61, refers to requirements established in "ASME OM Code - 1990," which is an older version of the OM code.

Please provide an exemption request for referencing an older issue of the ASME OM Code, or revise the statement to adopt the latest edition and addenda to the ASME OM Code incorporated by reference in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a.

<u>Series 261 - Initial Test Program</u>

261.011

Based on Revision 0 to the AP1000 DCD (redline/strikeout version comparing the AP600 and AP1000 DCDs), WCAP-13856, "AP600 Implementation of the Regulatory Treatment of Non-safety Related System Process," Revision 1, January 1998 has been deleted as a reference (this document is listed as Reference 6 in Section 14.3.9). SECY-95-132, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs," dated May 22, 1995, sets forth the Commission's policy with respect to those systems in passive light water reactors that are not designated as safety-related but may have a significant role to play in accident and consequence mitigation. Westinghouse has deleted reference to this document and does not mention the RTNSS process for the AP1000. Please justify why this information is being deleted from the DCD.

Please provide a discussion of how the AP1000 design certification application aligns with the Commission's policy on the RTNSS in passive plant designs as documented in SECY-95-132. Please discuss the process utilized to determine which defense-in-depth systems fall into the RTNSS category. Is WCAP-13856 applicable to AP1000? Please state if there are any similarities in the implementation of determining which systems fall into the RTNSS category for the AP1000 as compared to the AP600. If there are any differences please describe and discuss. Are there any changes in the systems that fall into the RTNSS category when comparing the AP1000 design to the AP600 design?

261.012

DCD, Tier 2, Section 14.2.9.1.1, "Containment Hydrogen Control Testing," contains general test acceptance criteria and methods for pre-operational testing of the passive autocatalytic recombiners (PARs) in the AP1000 design and Table 14.3-1, "ITAAC Screening Summary," lists an Inspection, Test, Analysis and Acceptance Criteria (ITAAC) for the Containment Hydrogen Control System. However, based on Revision 0 of the AP1000 DCD, reference and design feature information regarding PARS that was contained in the AP600 DCD has been deleted and is no longer present in AP1000 DCD Section 14.3, "Certified Design Material." Specifically, Westinghouse has deleted (1) the reference to Table 6.2.4.2 from Table 14.3-6, "Probabilistic Risk Assessment," and (2) the reference to Section 6.2.4.2.2 from Table 14.3-8, "Severe Accident Analysis." Westinghouse also removed the 3rd PAR in the in-containment refueling water storage tank (IRWST) from the AP1000 design. In the AP600 design, the third PAR is used to mitigate hydrogen combustion events that could possibly damage or prevent the IRWST from performing its intended safety function. (See RAI 480.001 for additional technical details.) Please provide additional information justifying deletion of the PARs from the AP1000 DCD.

261.013

In the AP1000 DCD, Tier 2, Section 14.3, Table 14.3-2, "Design Basis Accident Analysis," page 14.3-30, it states that "Nominal values for pertinent plant parameters used in accident analysis with 10% steam generator tube plugging - Reactor coolant flow (gpm) - 296,000 [as described in Section 15.03]." Table 5.1-3, "Thermal Hydraulic Parameters," states that the "Minimum Measured Flow (MMF), flow rate, gpm/loop, with 10% Tube Plugging" is 150,835 gallons-per-minute (gpm) or 301,670 gpm total reactor coolant flow. Table 15.0-3 lists the reactor coolant pump flow per loop with "Revised Thermal Design Procedure (RTDP) with 10% Steam Generator Tube Plugging" as 15.08 E+04 gpm which equals 301,600 gpm total reactor coolant flow. In DCD Chapter 16, "Technical Specifications," Section 3.4, "Reactor Coolant System," 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," LCO 3.4.1c, states "RCS total flow rate $\geq$[301,670] gpm." Based on these cross references to total reactor flow rates in the AP1000 design, the nominal total flow value with 10% steam generator (SG) tube plugging in Table 14.3-2 should be changed to 301,670 gpm.

Series 720 - Reliability and Risk Assessment

Reference: AP1000 Probabilistic Risk Assessment, APP-GW-GL-022, Revision 0, March 2002

720.027

The staff review identified several differences between the AP600 and the AP1000 probabilistic risk assessments (PRAs) in the assumed initiating event frequencies for several accident categories. These differences are related to (a) various LOCA categories, (b) steam generator tube rupture (SGTR) accidents, and (c) passive residual heat removal (PRHR) heat exchanger (HX) tube rupture accidents.

For the LOCA categories, these differences are primarily due to the selective use in the AP1000 PRA of operating experience data reported in NUREG/CR-5750, “Rates of Initiating Events at U.S. Nuclear Power Plants: 1987 - 1995,” February 1999, for pipe breaks as opposed to the use of data from pipe break analyses in the AP600 PRA. In the AP1000 PRA, operating experience data are used for some pipe break frequencies (e.g., pipe breaks contributing to medium and large LOCAs) but not for others (e.g., pipe breaks in the direct vessel injection lines). Similarly, the frequencies of contributors to the various LOCA initiating event categories other than pipe breaks, such as stuck-open pressurizer valves, were calculated in the AP1000 PRA while operating experience data are reported in NUREG/CR-5750 for these contributors.

The initiating event frequency for SGTR events is assumed in the AP1000 PRA to be 3.88E-3 events/year as opposed to 5.2E-3 events/year used in the AP600 and 7E-3 events/year reported in NUREG/CR-5750. In Attachment 2A, it is stated that the AP1000 SGTR frequency is based on 1.94E-7 failures per tube-year instead of 1.25E-6 failures per tube year used in the AP600 PRA because the AP1000 SG will be manufactured using Alloy 690 which is more resistant to stress corrosion cracking than the Alloy 600 assumed in the AP600 design. It is argued that due to this design improvement the SGTR frequency is smaller for the AP1000 design than the AP600 design even though the AP1000 SGs have many more tubes (i.e., 20,000 for the AP1000 versus 12,614 for the AP600). Please explain how the SGTR frequency per tube year for the AP1000 was calculated, including data and assumptions.

The initiating event frequency for PRHR HX tube rupture events is assumed in the AP1000 PRA to be 1.34E-4 events/year as opposed to 2.5E-4 events/year used in the AP600 PRA even though there are more and longer tubes in the AP1000 design. Statements made in Section 2.3.1.4 imply that the reliability of the AP1000 design PRHR HX tubes has improved because the reliability of the AP1000 SGTR tubes has improved with respect to the AP600 design. Please explain or re-calculate the PRHR HX tube rupture frequency to address these comments.

The frequency of PRHR HX tube rupture events is shown in the associated event tree (page 4-185) as 2.83E-4 events/year which is different from 1.34E-4 events/year reported in Chapter 2. Please explain this apparent discrepancy and revise appropriately.

The staff believes that a better understanding is needed of the impact of these new frequencies, including uncertainties, on the PRA results and insights which were derived from the AP600 PRA and assumed to be valid also for the AP1000. Examples of insights and conclusions that may need to be revised are the identification of low thermal margin but risk

significant success sequences for thermal-hydraulic (T-H) uncertainty assessment and the level of availability control for non-safety-related equipment. Please provide information addressing this issue.

720.028

The staff review identified two potentially significant differences between the AP600 and the AP1000 PRAs in the categorization of LOCA initiating events. One difference involves combining two AP600 PRA LOCA categories into one AP1000 PRA category. In the AP600 PRA, distinct medium LOCA (MLOCA) and an intermediate LOCA (NLOCA) categories were defined. The MLOCA involved break sizes between six and nine inches equivalent diameter and the NLOCA involved break sizes between two and six inches equivalent diameter. In the AP1000 PRA, however, these two categories were combined into one category (MLOCA which range from two to nine inches equivalent diameter). The other difference involves the splitting of one AP600 PRA LOCA category into two categories in the AP1000 PRA. In the AP600 PRA, the large LOCA category included both pipe breaks and spurious opening of the automatic depressurization system (ADS) valves. In the AP1000, there is one large LOCA category for pipe breaks and another large LOCA category for spurious ADS actuation.

T-H analyses performed for the AP600 design have identified bounding parameters (e.g., break size and location) for each LOCA category. The success criteria for systems and operator actions used in the AP600 PRA were determined by studying the plant response given such bounding parameters. For example, in the AP600 PRA, the success criteria for an MLOCA (i.e., six to nine inches equivalent diameter) are based on a cold leg break while for an NLOCA (i.e., two to six inches equivalent diameter) they are based on a hot leg break. These two AP600 LOCA categories which are associated with different plant responses were combined in the AP1000 PRA. Since the AP1000 PRA success criteria are based to a certain extent on information from analyses performed for the AP600 design, please explain the following:

A. Why these different initiating event categorizations were needed;

B. How information from AP600 design T-H analyses is used to support AP1000 design criteria; and

C. The approach that was followed, including new analyses beyond those performed for the AP600, to ensure that the AP1000 success criteria are valid for all break sizes and locations in each category and that the T-H uncertainties for risk-significant sequences are bounded.

720.029

In Section 6.3.2.5, the time windows available for several operator actions associated with specific LOCA sequences are discussed. They include operator actions to actuate the core makeup tanks (CMTs), depressurize the reactor coolant system (RCS), and actuate the normal residual heat removal (NRHR) pumps. In many cases, more than one time window (success criteria for operator action) is defined depending on the success or failure of other systems. For example, for medium LOCAs and CMT line breaks it was determined that the time available for operator action to manually actuate CMT injection (from the time the first signal would alert the operators to the time CMT actuation would occur) is 10 minutes without accumulator injection and 20 minutes with accumulator injection. Similarly, for medium LOCAs and CMT line

breaks it was determined that the time available for operator action to manually depressurize the RCS is 20 minutes for cases with successful accumulator injection and PRHR injection. This time of availability was assessed to be too short to take credit for operator action for cases without accumulator injection or PRHR operation. The success criteria for operator actions used in the AP1000 human reliability analysis (HRA) are summarized in Table 6.3. For the specific example of medium LOCAs and CMT line breaks, two operator actions are listed:
(1) manually actuate the CMTs if automatic actuation fails (event CMN-MAN01); and
(2) recognize the need for RCS depressurization (event LPM-MAN02). It appears from the HRA, documented in Chapter 30, that the human error probabilities for the events CMN-MAN01 and LPM-MAN02 were calculated assuming a 20-minute time window, independent of the success or failure of the accumulators and PRHR (actually, the PRHR is not modeled at all in the MLOCA and CMT line break event trees). Please explain how the various success criteria for operator actions (time windows), determined by T-H analyses of the various accident sequences, were modeled in the PRA.

720.030

The AP1000 PRA event trees include a top event for containment cooling (event CHR). In Chapter 4 (Event Tree Models) it is stated that this top event models the need to successfully remove thermal energy from the containment atmosphere to the environment via the containment vessel following events that cause a significant increase in containment pressure and temperature, such as a LOCA and main steam line break (MSLB) accident inside containment. However, several statements made in Chapter 6 (Success Criteria Analysis) imply that the passive containment cooling system (PCS) water is not needed to prevent core damage. For example, on page 6-8 it is stated: “Therefore, sequences in which core damage has been avoided with successful IRWST injection and recirculation represent success (i.e., no core damage) regardless of the status of containment integrity or PCS water.” Furthermore, on page 6-9 it is stated: “For success paths that result in steam release to the containment, the success of containment cooling (PCS or RNS) is modeled. If containment cooling is successful, then the path ends in an OK state. If PCS water cooling is not successful, then the path goes to a special OK end state to allow containment integrity sensitivity studies to be made.” This “special OK” end state is labeled “late containment failure (LCF)” end state on page 4-141 and defined as an end state “...where the containment heat removal by either passive containment cooling system (PCS) or component cooling water (CCS) heat exchangers via normal residual heat removal (RHR) fails.” These and other similar statements throughout the PRA create some confusion regarding what constitutes the PCS and what the criteria for its success are. In addition, no attempt is made to assess the impact of the “special OK end state” on PRA results and insights. Please address the following items:

A. Is containment cooling by heat transfer through the containment shell to the outside air included in the functions of the PCS system? Some parts of the PRA imply that containment cooling by air flow is part of the PCS while elsewhere in the PRA it is stated that the PCS is identical with the passive containment system water. If the answer is yes, please revise the system description and success criteria accordingly. Otherwise, include a statement at the beginning of each chapter discussing containment cooling, such as Chapter 4 (event tree models) and chapter 6 (success criteria), to clarify that air cooling is not part of PCS.

B. The success criteria for containment cooling are included in Section 6.3.1.5 under "Containment Isolation." Please include a separate section discussing the success criteria for "Containment Cooling."

C. In Chapter 13, on PCS, three fault tree models are listed. One, labeled PCT, is for all transients and LOCA events. Another, labeled PCP, is for loss of offsite power accidents. A third, labeled PCB, is for station blackout accidents. Table 13-2 summarizes the success criteria for fault tree PCT only. Since the PCS is a passive system, loss of offsite power or station blackout should not have an impact on the fault tree of this system. Please explain how fault trees PCP and PCB are different from PCT.

D. It is stated in Chapter 13 on PCS that blockage or plugging of the air flow paths is not modeled in the PRA because of the many design features that make such failure mechanisms highly unlikely (e.g., fifteen air flow path inlets covered by screens, each provided with a heating source to prevent blockage due to buildup of snow or ice). The staff believes that even "highly unlikely" failures should be included in the PRA models when such failures are associated with certain systems, structures and components (SSCs) whose unavailability or degradation would increase the plant's risk significantly (i.e., SSCs associated with high risk achievement worth). An SSC's risk achievement worth is used to identify operational requirements, such as requirements for surveillance and maintenance, to ensure that failure rates for these SSCs do not increase and remain "highly unlikely." In the case of containment air flow cooling paths, is air blockage highly unlikely when the availability of the heating sources to prevent buildup of snow or ice is not properly monitored and maintained? This issue may be more significant for the AP1000 than it is for the AP600 because of the higher power and, thus, lower thermal margin associated with the AP1000 as compared to the AP600. Please provide a discussion on how this issue is being addressed in the AP1000 PRA.

E. On page 6-9 it is stated that "If PCS water cooling is not successful, then the path goes to a special OK end state to allow containment integrity sensitivity studies to be made." This "special OK" end state is labeled "late containment failure (LCF)" end state on page 4-141. Please explain the reasons for considering containment integrity sensitivity studies but not core damage sensitivity studies. Would the assumed late containment failure not cause core damage and large release? Also, is containment cooling by air flow alone adequate for long-term operation of the PRHR system even in the presence of an open or failed containment? Please explain and provide supporting analyses, as necessary.

F. The success criteria for containment heat removal, as listed in Table 6-2 and associated footnote #17, indicates that heat transfer through the containment shell to the outside air is sufficient to cool the containment atmosphere even with an open containment. According to Table 6-2, the success criteria for containment cooling by air flow for all accident sequences are based on T-H analyses documented in Appendix A of the PRA. However, the staff was not able to locate such analyses. Please explain.

G. If there is significant uncertainty as to whether the "late containment failure (LCF)" end state can lead to core damage, please perform a sensitivity study to assess the potential impact of this uncertainty on PRA results and insights, including insights related to the identification of non-safety-related equipment for availability control as well as the level of availability control for such equipment.

720.031

The success criteria listed in Table 6-2 for the “containment integrity” function appears to apply to the “core cooling” function. Please explain and revise accordingly.

720.032

In Section 6.3.1, “General Sequence Success Criteria,” it is stated (page 6-3): “In general, if the reactor achieves a stable shutdown condition without core damage, and this condition can be maintained for at least 24 hours following event initiation without further action or system operation, the sequence is categorized as successful. It is not sufficient to avoid core damage during the first 24 hours if conditions have not stabilized and core damage is anticipated shortly following 24 hours. That is, core damage is assumed if the RCS conditions are not stabilized in 24 hours, or if core damage is anticipated following 24 hours without further system or operator action.” Please describe the accident sequences where the conditions are not stabilized in 24 hours or core damage is anticipated following 24 hours without further system or operator action.

720.033

Several statements and common cause failure (CCF) probabilities related to explosive (squib) valves, included in Chapter 12 on “Passive Core Cooling/In-Containment Refueling Water Storage Tank” and in Chapter 29 on “Common Cause Failure Analysis,” appear to conflict with each other. For example, in Section 12.6.1 (page 12-6) it is stated: “Common cause failures of the valves in the injection paths are modeled separately or independently of common cause failure of the valves in the recirculation paths.” However, in Section 29.3.1 (page 29-6, assumption no. 9) it is stated: “The IRWST injection system consists of four high-pressure (HP) explosive valves (EV). The IRWST recirculation system consists of two HP EVs and two low-pressure (LP) EVs. For this analysis, these valves are grouped into one common-cause failure group of six HP EVs and one common-cause failure group of two LP EVs.” Another example of conflicting statements is the definition of event IWX-EV3-SA which in Table 12-8 is defined as the CCF of two squib valves in the recirculation lines while in Table 29-2 it is defined as the CCF of three out of six HP EVs. Please clarify this apparent discrepancy and revise appropriately. In addition, please provide information to clarify the following items:

A. The staff was not able to find a description of the design and operational differences between HP and LP explosive (squib) valves. Please explain what makes these valves “diverse” so that CCF between them is considered negligible.

B. The calculation of CCF probabilities for several sets of explosive and check valves is documented in Chapter 29 and summarized in Table 29-2. Several discrepancies appear in the assessed probabilities for the various sets of valves. For example, the CCF of all six HP EVs (event IWX-EV-SA) was assessed to be 2.6E-5/demand while the CCF probability of only two HP EVs (event IWX-EV1-SA) was assessed to be 5.8E-6 per demand (i.e., smaller than the CCF probability of all six valves in both injection lines). Similarly, the CCF probability of all four injection check valves (event IWX-CV-AO) was assessed to be 3.0E-5/demand while the CCF probability of only the two check valves in one injection line (event IWX-CV1-AO) was assessed to be 5.4E-7/demand. It appears that this discrepancy resulted from an erroneous application of the methodology used to assess CCF probabilities. These CCF probabilities have a

significant impact on PRA results and insights as the importance analyses reported in Chapter 50 indicate. Please explain and revise as necessary.

C. The HP EVs were grouped into one common-cause failure group when both injection lines are available but not when only one injection line is available, as is the case of a direct vessel injection (DVI) line break accident. Please explain the basis for this apparent discrepancy and revise as necessary.

720.034

The sensitivity of the PRA results to the unavailability of the non-safety related standby systems was investigated and the results are summarized in section 50.5.4 (sensitivity case no. 36). However, two different core damage frequency (CDF) numbers are reported in page 50-14. A CDF of 2.92E-5/year is reported at the end of Section 50.5.4 while in Section 50.6 (Results) a CDF of 7.345E-6/year is reported. Please explain and revise as necessary.

720.035

In Chapter 26 on the "Protection and Safety Monitoring System (PMS)," in Chapter 27 on the "Diverse Actuation System (DAS)," and in Chapter 28 on the "Plant Control System (PLS)," the following statement is made: "Because of the rapid changes that are taking place in the digital computer and graphic display technologies employed in the modern human systems interface, design certification of the AP1000 focuses upon the process used to design and implement instrumentation and control systems for the AP1000, rather than on the specific implementation." To be able to take advantage of such changes in technology, design options in addition to the ones used in the AP600 design certification are proposed for the safety-related PMS and the non-safety related DAS and PLS. For the safety-related PMS, the option to use the Common Qualified Platform (Common Q) is proposed. For the non-safety-related DAS and PLS, the option to use commercial off-the-shelf hardware and software which will be current at the time of construction is proposed. Please provide more detailed information regarding the implementation of the proposed options by responding to the following questions:

A. Regarding the PMS, it is stated that the AP600 Instrumentation and Control (I&C) functional requirements which have received design certification will be retained to the maximum extent compatible with the Common Q hardware and software. Also, it is stated that although the details of the AP1000 PRA model follow the AP600 design, the Common Q hardware and software provide a degree of redundancy that is equivalent to the redundancy modeled in the AP1000 PRA. Please explain the process that will be used to verify that a PMS designed with the "Common Q" option will have equivalent or better reliability with the system modeled in the PRA. Also, please explain how the introduction of the "Common Q" option will affect important PRA-based insights about the PMS, such as the ones identified during the AP600 PRA review (i.e., the design certification information "PRA-based insights" documented in Table 19.59-29 of the AP600 DCD).

B. Regarding the non-safety-related DAS and PLS, it is stated that the AP1000 PRA is based on "one possible ....configuration designed to meet the requirements of DCD Chapter 7" and that "the functional requirements and the degree of redundancy modeled in the PRA are representative of the expected final ....design." Please explain the process that will be used to verify that the DAS and PLS, designed with the "commercial off-the-shelf hardware and

software current at the time of construction" option will have equivalent or better reliability with the systems modeled in the AP1000 PRA. Also, please explain how the introduction of such an option will affect important PRA-based insights about the DAS and PLS, such as the ones identified during the AP600 PRA review (i.e., the design certification information "PRA-based insights" documented in Table 19.59-29 of the AP600 DCD).

720.036

In Appendix A (page A-29), several accident sequences which were identified in the AP600 PRA as "low T/H margin, risk important" scenarios, are discussed with respect to their potential applicability to the AP1000 design. While the first three cases discussed small LOCAs (SLOCAs), Appendix A states that "The PRHR HX is included for the AP1000 because the success criteria has been changed to require PRHR HX operation for MLOCAs with failure of CMTs." Please clarify and revise accordingly.

720.037

In Appendix A (pages A-28 and A-29) it is stated that a process to evaluate T/H uncertainty was developed for the AP600 design certification. This process included the identification of "low T/H margin PRA important accident scenarios" which were analyzed with design-basis accident analysis computer codes to bound the T/H uncertainty. It is argued that no additional sequences can be classified as "low T/H margin PRA important accident scenarios" for the AP1000 beyond those identified and analyzed for the AP600 design because the AP1000 plant/systems are very similar in design and capability to the AP600's and the PRA results are similar. The staff believes that additional analysis is needed to support this argument. Even though the two designs use similar systems with comparable capabilities to respond to design-basis accidents, differences in sizes, flow rates, heat transfer areas and decay heat production exist between the two designs. The impact of such differences to plant response can be significant for beyond design-basis accident scenarios involving multiple failures and, potentially, system interactions. In addition, changes in initiating event categories, initiating event frequencies, and success criteria in the AP1000 PRA, as compared to the AP600 PRA, could have increased the risk significance of accident scenarios. Please use a systematic approach to identify "risk significant low margin" sequences for detailed T-H uncertainty assessment and provide additional information addressing this issue.

720.038

An important objective of the AP600 design certification PRA was to identify important PRA insights and assumptions and make sure that they have been addressed in the design certification through design certification requirements, such as requirements for inspection, tests, analyses and acceptance criteria (ITAAC), the requirement for a design reliability assurance program (D-RAP), and combined operating license (COL) action items. These requirements were incorporated in the DCD, Table 19.59-29 "PRA-based insights," to ensure that any future plant which references the design will be built and operated in a manner that is consistent with important assumptions made in the design certification PRA. Please provide similar information for the AP1000. Since the major part of this information is expected to be the same as for the AP600, please start with DCD Table 19.59-29 and highlight the differences in "insights" between the two designs.

720.039

An important objective of the AP600 PRA was to provide input to the design certification process regarding the need for regulatory oversight of certain non-safety-related systems. The process used to identify SSCs for regulatory oversight as well as the type and level of such oversight was called Regulatory Treatment of Non-Safety-Related Systems (RTNSS) in the AP600 certification. The end results of this process were the “availability controls” documented in Section 16.3 of the DCD. Please provide similar information for the AP1000. This information should account for uncertainties in the AP1000 PRA so that it can be used by the staff to make similar conclusions, about the need for non-safety-system oversight, as were made for the AP600 design (e.g., as documented in the AP600 Final Safety Evaluation Report (FSER) Chapter 19.1.7 “PRA input to the RTNSS Process.)”

720.040

Please provide representative examples of PRA used in the AP1000 design process to achieve each of the following objectives: (a) enhance the AP1000 design by adding or modifying design features or operational requirements; (b) quantify the effect of new design features and operational strategies on plant risk to confirm the risk reduction credit for such improvements; (c) select among alternative features, operational strategies or design options.

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	ML022560265
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		

Letter No.	Date issued	ADAMS Accession No.	RAI Nos.	Date of response	ADAMS Accession No.
9	9/24/02	ML022620079	250.001-250.003, 251.001-251.029, 252.001-252.009, 281.001-281.003		

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