September 18, 2002

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

SUBJECT: RE-ISSUANCE OF REQUEST FOR ADDITIONAL INFORMATION LETTER NO. 6 - AP1000 DESIGN CERTIFICATION REVIEW (TAC NO. MB4683)

Dear Mr. Corletti:

The U. S. Nuclear Regulatory Commission staff issued Request for Additional Information (RAI) Letter No. 6 associated with the reactor systems and probabilistic risk assessment success criteria portions of the AP1000 design on September 5, 2002. Since issuance of this document, the staff discovered typographical errors in some of the questions. Although these errors should not impact your efforts in responding to these RAIs, we would like to correct these errors. This correspondence replaces the previous RAI Letter No. 6 in its entirety.

A correct version of these RAIs was sent to you via electronic mail on August 22 and August 26, 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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Request for Additional Information (RAI)
AP1000 Standard Design Certification
Series 440 - Reactor Systems and
Series 720 - Reliability and Risk Assessment

SERIES 440 - REACTOR SYSTEMS

The following questions are related to AP1000 Tier 2 Information:

440.009

Provide steady-state pressure drops through the vessel and primary loops of the AP1000 design at best estimate flow conditions for a 10 percent steam generator (SG) tube plugging level, including the non-recoverable pressure loss through the following regions and components:

A. Vessel inlet nozzle
B. Lower plenum
C. Lower core plate and fuel assembly inlet nozzle
D. Core
E. Upper core plate and fuel assembly outlet nozzle
F. Upper plenum
G. Outlet nozzle
H. Hot leg
I. Steam generator, including inlet and outlet plena
J. Cold leg

440.010

For the AP1000 14-foot core design, the 14-foot fuel assemblies tend to have more vibrational problems than the 12-foot fuel assemblies in the currently operating power water reactors (PWRs).

Provide operational experiences, test programs, and/or surveillance plans that demonstrate the flow vibration will not present a serious operational problem to the AP1000 core.

440.011

Provide technical justifications demonstrating that the Nuclear Regulatory Commission (NRC) approved neutronics and thermal-hydraulics computer codes such as TWINKLE and ANC, are valid for application in analyzing the AP1000.

440.012

Tier 2 Information Section 4.1 states that “other types of fuel may be used ....” The staff assumes that these “other types” of fuel will be NRC-approved fuel types.

Is this a correct assumption?

440.013

Section 4.3.1.7, on anticipated transient without scram (ATWS), makes reference to topical report WCAP-11992, “Joint Westinghouse Owners Group/Westinghouse Program: ATWS Rule Administrative Process,” dated December 1988, which was rejected by the NRC staff (letter from NRC to Vanderburg, Chairman, Westinghouse Owners Group, “Review of WCAP-11922,” dated July 1, 1997).

Since this topical report is not acceptable for referencing in licensing applications, it should be removed from this submittal.

440.014

Provide quantitative technical analysis to support that the AP1000 can meet the present ATWS Rule that requires an ATWS mitigation system actuation circuitry (AMSAC), which automatically initiates the auxiliary feedwater system and initiates a turbine trip under conditions indicative of an ATWS, and meets the basis of the rule, i.e., the reactor vessel pressure will exceed 3200 pounds-per-square inch (absolute) (psia) for no more than five percent of the cycle time. The analysis should be performed for all applicable non-loss-of-coolant accident (LOCA) transients in order to identify the limiting ATWS case. Discuss the methods used and verify that the methods are acceptable. Also, justify that the assumptions for the applicable ATWS analyses are adequate as they relate to input parameters such as the initial power level, moderator temperature coefficient (MTC), pressurizer safety and relief valves capacity, reactor coolant system (RCS) volume, SG pressure, passive residual heat removal system (PRHR) heat transfer capacity and its actuation delay time, and the AMSAC setpoint to trip the turbine and initiate the PRHR. Also include a discussion and applicable values of the unfavorable exposure time for the MTC (in accordance with the NUREG-0460, “Anticipated Transients Without Scram for Light Water Reactors,” guidance for the newer plant design) assumed in the analyses.

440.015

The second paragraph of Section 4.3.2.3.2.1, “Moderator Density and Temperature Coefficients,” makes reference to the possibility of a positive net value for the moderator coefficient. The third paragraph of this same section states that the value of moderator temperature coefficient (MTC) over the range of power operation is negative.

A. Provide values of the MTC in tabular form or graphical form for values of the moderator coefficients at Hot Zero Power (HZP) and at Hot Full Power (HFP) as a function of core life for an expected typical cycle of a AP1000 core.

B. If indeed the MTC is positive, how will it impact the Technical Specification (TS) associated with the MTC?

C. If the MTC is positive, how does it impact the ATWS event with regards to exceeding the vessel pressure of 3200 psia and the amount of time the core is in an “Unfavorable Exposure Time (UET)”?

D. The UET domain is a domain where the vessel pressure is known to be exceeded. Please provide the range of vessel pressures for the total time the core is in this domain, or as a function of core life for an expected typical cycle of a AP1000 core.

440.016

In Section 4.3.2.2.4, the fifth sentence in the second paragraph "The calibrated difference in power ...." is not a complete sentence. Please clarify.

440.017

Section 4.3.2.2.6 makes reference to topical reports WCAP-7811, "Power Distribution Control of Westinghouse Pressurized Water Reactors," WCAP-8385, "Power Distribution Control and Load Following," and WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control, F_Q Surveillance Technical Specification."

A. Both WCAP-7811 and WCAP-8385, which are used for determining power peaking factors, radial and axial distributions etc., are not approved by the NRC. Please provide technical justification for the use of these topical reports in analyzing the AP1000 reactor.

B. WCAP-10216-P-A was developed specifically for use in 12-foot cores only. Please provide technical justification to support the use of this topical report in the AP1000 14-foot core design.

440.018

Section 4.3.2.2.6 states that allowing for fuel densification effects, the average linear power at 3400 megawatts (MW) is 5.72 kilowatts-per-foot (kW/ft), and that from Figure 4.3-14, the conservative upper bound value of normalized local power density, including uncertainty allowances, is 2.60 corresponding to a peak linear heat rate of 15.0 kW/ft at each core elevation at 101 percent power. Since the AP1000 14-foot core design has a total of 41,448 fuel rods, which is equivalent to an average linear power of 5.86 kW/ft without fuel densification effects, it is not clear to the staff how the numbers stated in Section 4.3.2.2.6 are derived.

Provide additional supporting information to clarify the above statement.

440.019

Section 4.3.2.2.6 indicates the uncertainty factors for the hot channel heat flux factor, $F_{Q,}$ to be 1.03, 1.05, and 1.056, for the engineering uncertainty, method uncertainty, and rod bow effects, respectively.

Provide a derivation on the determination of these uncertainty factors values.

440.020

Section 4.3.2.3 states that the reactivity coefficients are calculated with approved nuclear methods. Please provide reference to these approved methods.

440.021

Tier 2 Information Section 4.4.4.5.2 cites WCAP-14565-P-A for the NRC approval of the VIPRE-W core model, and references (Reference 83) WCAP-14565-P-A with the title of "VIPRE-W Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis." However, the formal title of WCAP-14565-P-A topical report has the word "VIPRE-01" instead of "VIPRE-W."

Confirm that the VIPRE-W code is the same version of VIPRE-01 code discussed in the WCAP-14565-P-A topical report. Discuss whether Westinghouse intends to change the word in the WCAP-14565-P-A topical report from "VIPRE-01" to "VIPRE-W."

440.022

Section 4.4.1.1.2 states that for those transients that use the VIPRE-W computer program and the WRB-2M correlation, the Revised Thermal Design Procedure (RTDP) design limits are 1.25 for the typical cell and 1.25 for the thimble cell for core and axial offset limits, and 1.22 for the typical cell and 1.21 for the thimble cell for all other RTDP transients, and that these values may be revised when plant specific uncertainties are available.

A. Discuss the differences between the RTDP design departure from nucleate boiling ratio (DNBR) limits for (1) core and axial offset limits, and (2) other RTDP transients, respectively.

B. Provide the derivations of these RTDP design DNBR limits, including the uncertainties of all parameters used in the derivation.

C. Provide the instrument uncertainty methodology and the assumed uncertainty values of various components of the instrument for the measurements of the parameters included in the RTDP.

440.023

In Tier 2 Information, Section 4.1 states that (1) the AP1000 fuel assemblies are the same as the 17x17 XL Robust fuel assemblies except that they have four intermediate flow mixing grids in the top mixing vane grid span, (2) the XL Robust fuel assembly evolved from VANTAGE 5 HYBRID design, and (3) the AP1000 fuel assembly design also includes a protective grid for enhanced debris resistance.

Since the WRB-2M critical heat flux correlation is developed from test assemblies designed to simulate modified VANTAGE 5 H fuel with or without modified flow mixer grids, provide justification why these test data and the WRB-2M correlation are applicable to the AP1000 fuel design. The justification should discuss the effects on the critical heat flux of (1) differences in the modified intermediate flow mixer grids in the test assemblies and the intermediate flow mixer grids in the AP1000 fuel design, and (2) any other differences of the test assemblies from the AP1000 fuel design, such as the protective grid.

440.024

Section 4.4.2.2.1 shows the applicable ranges of various parameters of the WRB-2M and WRB-2 correlations, respectively, including the local mass velocity applicability range from 0.97 to 3.1M-lb/ft^2-hr for WRB-2M, and between 0.9 and 3.7M-lb/ft^2-hr for WRB-2.

A. Are there checks in the VIPRE-W code to verify that these correlations are not used outside their applicability ranges? If not, how is it ensured that the correlations are applied within their applicability range?

B. Will the local mass velocity for all design-basis transients ever fall below the lower bound applicability ranges of these correlations? If so, which critical heat flux correlation will be used for the low flow conditions, and what is the basis for its applicability to the AP1000 fuel design?

440.025

Section 4.4.2.2.1 states that the WRB-2 or W-3 correlation is used wherever the WRB-2M correlation is not applicable; and that the W-3 correlation is used in the heated region below the first mixing vane grid and in the analysis of accident conditions where the system pressure is below the range of the primary correlation.

A. Are both the WRB-2 and W-3 correlations applicable to the AP1000 RFA [robust fuel assembly] XL fuel design?

B. Describe under what conditions the WRB-2 correlation will be used in place of WRB-2M and how this is done in the VIPRE-W code.

C. Do the test assemblies used for the development of the WRB-2 correlation have the same grid design as that of the AP1000 RFA XL grid design?

D. Clarify, and explain, if the W-3 correlation is used in the heated region below the first mixing vane for all thermal-hydraulic conditions where the WRB-2M and WRB-2 correlations are applicable, and how this is done in the VIPRE-W code.

E. Since the WRB-2M correlation is used in combination with the RTDP methodology, will the use of the WRB-2 or W-3 correlation in place of WRB-2M be applied in the same fashion? What are the RTDP DNBR limits for core and axial offset limits and for RTDP transients for the WRB-2 and W-3 correlations, respectively? Provide derivations of these limits.

440.026

Section 4.4.2.2.4 states that the effects of variations in pellet diameter, density, and enrichment on enthalpy rise engineering hot channel factor are considered statistically in establishing the limit departure from nucleate boiling ratios (DNBRs) for the Revised Thermal Design Procedure, and that uncertainties in these variables are determined from sampling of manufacturing data.

Provide the assumed uncertainty values, and bases, assigned in the RTDP for the variations in pellet diameter, density, and enrichment of the AP1000 RFA XL fuel design.

440.027

Section 4.4.4.3.2 states that the minimum DNBR is calculated for the design power shape for non-overpower/overtemperature departure from nucleate boiling (DNB) events, and this design shape results in calculated DNBR that bounds the normal operation shapes.

Explain the design power shape for non-overpower/overtemperature events and how this shape bounds the normal operation shapes for the DNBR calculations.

440.028

Section 4.4.4.5.2 states that the effect of crud on the flow and enthalpy distribution in the core is not directly accounted for in the VIPRE-W evaluations; however, conservative treatment by the VIPRE-W modeling method has been demonstrated to bound this effect in DNBR calculations [WCAP-14565-P-A].

Provide analysis or data that demonstrates how the VIPRE-W modeling method bounds the effect of crud on the flow and enthalpy distribution.

440.029

Section 4.4.1.2.1 states that the NRC has approved the fuel design evaluations up to 60,000 megawatt-days-per-megatons uranium (MWD/MTU) in Reference 81 [WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report"] and up to 62,000 MWD/MTU in Reference 9 [WCAP-12488-A, "Westinghouse Fuel Criteria Evaluation Process"]. There is a footnote under Reference 9 that states "NRC Staff Approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5."

Explain the footnote under Reference 9.

440.030

Section 4.4.1.2.2 states that for higher burnups, the peak kilowatt-per-foot (kw/ft) experienced during Condition I and III events is limited to that maximum value which is sufficient to provide that the fuel center-line temperatures remain below the melting temperature of the fuel rods.

Explain why the statement is limited to only Condition I and III events.

440.031

Section 4.4.1.3 states that core cooling evaluations are based on the thermal flow rate entering the reactor vessel, and that a typical maximum value of 5.9 percent is allotted as bypass flow. Section 4.4.1.3.2 states that the maximum bypass flow fraction of 5.9 percent assumes the use of thimble plugging devices in the rod cluster control guide tubes that do not contain any other core components. Table 5.1-3 lists five components which constitute the 5.9 percent core bypass flow.

Describe how these flow fractions for the 5 components are determined.

440.032

Section 4.4.2.9.2 states that core and vessel pressure drops based on the best-estimate flow are quoted in Table 4.4-1, and that the uncertainties quoted are based on the uncertainties in both the test results and the analytical extension of these values to the reactor application. Table 4.4-1 lists the core and vessel pressure drops uncertainty values for the AP600 and Typical XL plant, but not for the AP1000.

Explain why the pressure drop uncertainties are not included for the AP1000, or provide an update of Table 4.4-1 to include the uncertainties for the AP1000 core and vessel pressure drops, similar to those shown for the AP600 and Typical XL Plant.

440.033

Section 4.4.3.1 states that total RCS volume including pressurizer and surge line and RCS liquid volume, including pressurizer water at steady state power conditions, are given in Table 5.1-2; and that the steady-state pressure drops and temperature distributions throughout the RCS are presented in Table 5.1-3. However, most of these information are not included in these tables.

Provide an update of Tables 5.1-2 and 5.1-3 to include these data.

440.034

Section 4.4.3.6 states that the thermal and hydraulic characteristics are given in Tables 4.1-1, 4.4-1, and 4.4-2. As shown in Table 5.1-3, certain thermal design parameters, such as the reactor coolant flow rate and temperatures, are affected by the amount of plugging of the SG tubes.

Update Tables 4.1-1 and 4.4.1 to clarify the percent of tube plugging for which the thermal and hydraulic design parameters are referring to.

440.035

In Tier 2 Information, Section 5.2.2.1 states that the sizing of the pressurizer safety valves (for overpressure protection of the RCS during power operation) is based on the analysis of a complete loss of steam flow to the turbine, with the reactor operating at 102 percent of rated power.

Provide the safety analysis to demonstrate that the relieving capacity and set pressure of the pressurizer safety valves specified in Table 5.4-17 are appropriate for overpressure protection with sufficient margin to account for uncertainties in the design and operation of the plant. The description should include the analysis methodology, assumptions, uncertainties in the design and operation of the plant, and the analysis results.

440.036

Section 5.2.2.1 states that a relief valve in the residual heat removal system (RNS) provides low-temperature overpressure protection (LTOP) for the RCS, and that the valve is sized to prevent overpressure based on the following design basis events with a water solid pressurizer: (1) the limiting mass input event of the makeup/letdown flow mismatch, and (2) the limiting heat input event of inadvertent start of a reactor coolant pump (RCP).

Provide the safety analyses of both the limiting mass-input and heat-input overpressure events to support the adequacy of the RNS relief valve relieving capacity and set pressure specified in Table 5.4-17 for the LTOP. The description should include:

A. The applicable RCS pressure-temperature limits (LCO 3.4.3) with corresponding neutron fluence values of the reactor vessel, or the effective full power years.

B. The analysis methodology and assumptions, including consideration of limiting single failure assumption, the instrumentation uncertainties of pressure and temperature measurements, the relief valve set pressure and accumulation, the dynamic head effect of the reactor coolant flow, and the static head between the pressure tap and the limiting vessel locations, and pressure overshoot.

C. The analysis results.

D. The determination of the LTOP enable temperature of 275°F TS LCO 3.4.15.

440.037

In Tier 2 Information, Section 5.3.2.6.1.2 discusses the least squares adjustment procedure proposed to be applied in the dosimetry evaluation. The estimate of the uncertainties associated with the dosimeter activation measurement involves use of variances and co-variances. If a code which has not been approved by the staff is used, then the values of the variances and co-variances should be listed and their applicability to the AP1000 justified.

440.038

Section 5.3.4.1, on reactor vessel integrity and design states that "... reactor vessel is designed and fabricated ... to General Design Criteria 1, General Design Criteria 30 and 50.55a..." You should also state that General Design Criteria 14 and 31 are also applicable in the pressure vessel design.

440.039

In Tier 2 information, Table 5.4-1 provides the AP1000 canned-motor RCP design parameters, including the design flow, developed head, and motor/pump rotor moment of inertia.

Provide an AP1000 canned-motor pump head-capacity design value.

440.040

Some of the AP1000 RCP design parameters listed in Table 5.4-1 are different from the values provided in the Westinghouse presentation of May 9, 2002, in the NRC headquarters (see meeting summary dated May 9, 2002). These include unit overall weight, total weight, pump developed head, and motor/pump rotor moment of inertia.

Clarify which are the correct values.

440.041

AP1000 TS Bases B 3.4.4 describes that the AP1000 RCPs are powered by variable speed controllers when the reactor coolant temperature is below 500^{o}F, that the pumps must be started using the variable speed controller with the reactor trip breakers open, and that the controller shall be bypassed prior to closure of the reactor trip breakers. There is no discussion of the RCP and controller operation in Tier 2 Section 5.4.1, except that Table 5.4-1 specifies the RCP motor currents for starting, and nominal input, cold reactor coolant.

Provide a detailed description on the design and operation of the variable current RCPs and the variable speed controller, and modify Section 5.4.1, as necessary.

440.042

Section 5.4.1.2.1 states that the two canned-motor pumps that are directly connected to the two outlet nozzles on each SG channel head turn in the same direction.

Explain why the AP1000 design changes the turning direction of the canned-motor pumps from the AP600 design, where the two pumps turn in opposite direction.

440.043

Section 5.4.2.1 states that Chapter 15 discusses the accident analysis of a SG tube rupture, which is based on a rupture of one tube.

In SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” dated April 12, 1993, the NRC staff states its position that an applicant for a passive pressurized-water reactor (PWRs) design certification should assess features to mitigate the amount of containment bypass leakage that could result from the rupture of multiple SG tubes. This position arises from a concern that an multiple-tube rupture event creates the likelihood of a steam generator safety valve (SGSV) lifting and then failing to close, resulting in an unisolable release to the environment bypassing the containment.

A. Discuss the AP1000 design features that mitigate or prevent SGSV challenges during an event of rupture of multiple SG tubes.

B. Provide an analysis of multiple-tube rupture events to address the concern of containment bypass leakage resulting from a potential failure of the SGSV to reclose.

440.044

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.34(f)(2)(xiii), regarding Three Mile Island (TMI) Action Item II.E.3.1, requires that the pressurizer heaters be provided with a sufficient power supply and associated motive and control power interfaces to establish and maintain natural circulation in hot standby conditions with only on-site power available. Section 5.4.5.3.1 states that the AP1000 design conforms to this requirement because the pressurizer heater buses can be powered from the on-site diesel generators via manual alignment with sufficient capability to establish and maintain natural circulation in hot standby conditions. It also states that natural circulation cooling for a loss of all alternating-current (ac) power is a design basis for the AP1000, and that under a loss of all ac power, cooling is provided by the passive residual heat removal system (PRHRS). Since the AP1000 diesel generators are of non-safety grade design, the staff considers that the PRHRS with proper capability may provide an alternative to meet the intent of 10 CFR 50.34(f)(2)(xiii) to ensure availability of a safety grade decay heat removal method.

Provide analyses and/or test data to demonstrate and confirm the capability and reliability of the AP1000 PRHRS to maintain natural circulation cooling in hot standby conditions without the pressurizer heater power supply.

440.045

Section 5.4.7.2.2 describes the AP1000 normal residual heat removal system (RNS) design features addressing intersystem LOCA issue described in SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," dated January 12, 1990. Also, Section 1.9.5.1.7 addresses AP1000's compliance with the NRC position regarding the inter-system LOCA issue. It states that the AP1000 has similar fluid system design to the AP600; therefore, the conclusions of topical report WCAP-14425, "Evaluation of the AP600 Conformance to Inter-System Loss-of-Coolant Accident Acceptance Criteria," dated July 1995, are applicable to the AP1000.

Identify design differences between the AP1000 and the AP600, in terms of the design and design pressure of the primary or secondary systems and subsystems that directly or indirectly interface the RCS, that could affect the inter-system LOCA conclusions. For each of these differences identified, justify why the conclusions of WCAP-14425 are applicable to the AP1000.

440.046

Section 5.4.7 indicates that one of the major functions of the RNS is to provide a flow path for long-term post-accident makeup to the containment inventory. Section 5.4.7.1.1 states that this function is a safety-related function and safety design basis, whereas Section 5.4.7.1.2.6 lists this safety-related function under non-safety design bases.

A. Clarify the apparent discrepancy in the categorization of RNS long-term, post-accident containment inventory makeup path function.

B. Describe the RNS design features that satisfy this safety-related function.

440.047

Figure 5.4-6 shows that the relief valve on the suction line of the RNS is normally open and discharges into the containment sump, whereas Figure 5.4-7 shows the same valve to be normally closed and Section 5.4.9.2 states that the discharge from that valve is directed into the containment atmosphere.

Explain and correct the discrepancy.

440.048

Section 5.4.12.1 states that the reactor vessel head vent system (RVHVS) is designed to vent a volume of hydrogen at system pressure and temperature equivalent to approximately 40 percent of the RCS volume in one hour.

A. Describe the rationale for the appropriateness of this vent capacity design basis.

B. Do the AP1000 RVHVS valves have individual positive valve position indication and alarm in the control room?

440.049

Section 5.4.12 of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," specifies that procedures should be developed for use of the vent paths to remove gases that may inhibit core cooling from the U-tubes of the SGs; and that the procedures to operate the vent system should consider when venting is needed, and when it is not needed, with consideration of a variety of initial conditions, operator actions, and necessary instrumentation.

Describe the AP1000 procedures for venting the RCS system, including the criteria for opening and closing the RVHVS valves and automatic depressurization system (ADS) first stage valves, respectively, the bases for these criteria, the necessary instrumentation, and the procedures.

440.050

In Tier 2 Information, Section 6.3 describes the passive core cooling system. Each of the core makeup tank (CMT) and accumulator outlet injection lines contains a flow-tuning orifice that provides a mechanism for the field adjustment of the injection line resistance to establish the required flow rates assumed in the design. In the AP600 design, flow tuning orifices are also included in the in-containment refueling water storage tank (IRWST) injection lines.

Provide the reason why the AP1000 IRWST injection lines do not have flow-tuning orifices.

440.051

In Tier 2 Information, Table 6.3-3 indicates that the containment recirculation line motor-operated valves (MOV), V117A/B, are normally open. Figures 6.3-2 and 6.3-3 also show these MOVs are normally open. However, Section 6.3.2.1.3 states that "when the in-containment refueling water storage tank level decreases to a low level, the containment recirculation motor-operated valve and squib valves automatically open to ...," and "... In addition, the motor-operated valve path can be manually open to intentionally drain the in-containment refueling water storage tank to the reactor cavity during severe accidents." These descriptions appear to suggest the MOVs in the recirculation paths are normally closed.

Clarify the apparent discrepancy between Section 6.3.2.1.3 and the related table and figures, and make corrections, if necessary.

440.052

Section 7.3.1.2.4 describes the actuation logic and "preset time delays" of various stages of the ADS. For example, it states the preset time delay after the CMT Low-1 level setpoint (coincident with the CMT injection signal) for actuation of ADS, stage 1 (ADS-1), subsequent time delays for actuations of ADS, stage 2 (ADS-2) and ADS, stage 3 (ADS-3), and the preset time delay after the CMT Low-2 level setpoint (coincident with low RCS pressure). It also states the preset time delay between the actuation of the isolation valves and the depressurization valves for various stages for the ADS. Table 15.6.5-7 specifies the earliest actuation times (or delay times) for the actuations of various stages of ADS.

Where are the delay times between the actuation of the isolation valves and the depressurization valves for various stages of ADS specified? What are the values assumed in the Chapter 15 design-basis analyses? Do the earliest actuation times for various stages listed on Table 15.6.5-7 include the delay times between the actuation of the isolation and control valves?

440.053

Section 6.3.2.5.2 states that the passive core cooling system can sustain a single passive failure during the long-term phase and still retain an intact flow path to the core to supply sufficient flow to keep the core covered and to remove decay heat.

Describe your AP1000-specific analysis to confirm this conclusion.

440.054

Regarding Tier 2 Information, Chapter 15, “Accident Analysis,” and Appendix 19E, “Shutdown Evaluation,”

A. Provide a list of the methodologies and computer codes used in the LOCA and non-LOCA transient analyses and Appendix 19E shutdown evaluation for the AP1000 design certification application and reference the associated NRC acceptance letters to confirm the acceptance of the methodologies and codes used in the safety analyses.

B. Address the compliance with each of applicable limitations regarding the methodologies and codes and verify that the fuel performance, nuclear physics and thermal-hydraulics conditions of the analyses are within the applicable ranges of the approved computer codes.

440.055

Table 15.0-2, listing a summary of initial conditions and computer codes used in the accident analysis, is incomplete and inconsistent with the safety analyses in various sections of Chapter 15. For example, Section 15.1.2.2.1 indicates that the computer codes used for an increased feedwater event are LOFTRAN, FACTRAN and VIPRE, while Table 15.0-2 lists only LOFTRAN as the code used for the analysis. Section 15.4.8.2 indicates that for the rod ejection accident analysis, three codes are used: TWINKLE for the calculations of physics parameters; FACTRAN for the fuel rod temperature calculations, and THINC for the DNBR calculations. The use of THINC is neither discussed in Section 15.0-11, “Computer Codes Used,” nor included in Table 15.0-2. Also, Table 15.0-2 does not include information for the increase in reactor coolant inventory due to chemical and volume control system (CVS) malfunction event that is analyzed and discussed in Section 15.5.2.

Verify the accuracy of the information provided in Table 15.0-2 and revise the table if necessary to be consistent with applicable sections of Chapter 15.

440.056

Table 15.0-4a lists protection and safety monitoring system setpoints assumed in the accident analysis. These values for the setpoints are included in Table 3.3.1-1 of the TS as trip setpoints for the RTS. However, no total uncertainty allowances are specified for the reactor trips in the TS.

Address the acceptability of the TS for the trip setpoints without inclusion of the instrumentation uncertainties. This question on the TS setpoint uncertainties is also applied to TS Table 3.3.2-1, which specifies the trip setpoints for the engineered safeguards actuation systems without inclusion of the total uncertainty allowances.

440.057

Provide a list of the setpoints with the associated uncertainties for normal operation and the setpoints assumed in the transient analysis for engineered safety feature actuation systems, pressurizer safety valves, SG power-operated relief valves (PORVs) and safety valves.

Compare these analytical values with the applicable TS values and address the acceptability of the TS values.

440.058

Table 15.0-4a indicates that the setpoints assumed in the analysis for the safeguards (“S”) signal and steam line isolation on low steam line pressure are 405 psia and 535 psia for the cases with and without an adverse environment assumed, respectively.

A. Identify specific transients that rely on these low pressure signals (at both conditions with and without an assumed adverse environment) for consequence mitigation and specify the assumed conditions (such as temperature, pressure, moisture and radiation levels) for which the low pressure signal actuation setpoints are applicable.

B. Provide a discussion of the results of transient analysis and demonstrate that the analytical results are within the applicable range of the low pressure actuation signal setpoints.

C. Provide a TS that correctly reflects the setpoints of these low steam line pressure actuation signals with inclusion of the total allowance for measurement uncertainties.

440.059

Table 15.0-6 lists plant systems and equipment that are available for transient and accident conditions. The table is incomplete and inconsistent with the information provided in the various sections of Chapter 15. For example, it does not include (1) an increase in RCS inventory due to CVS malfunction events, and (2) small line breaks outside containment event, which are analyzed and discussed in Sections 15.5.2 and 15.6.2, respectively. Table 15.0-6 indicates that for the loss of external load and turbine trip events, the reactor trips available for the consequence mitigation are the trip signals from high pressurizer pressure, overtemperature delta T, overpower delta T, and the manual trip signal. The information is inconsistent with Section 15.2.3 (page 15.2.6), which indicates that for the turbine trip event, trip signals are expected due to high pressurizer pressure, overtemperature delta T, low RCP speed, high pressurizer water level, and low SG water level. The inconsistent information related to the available trip functions, engineered safety feature (ESF) actuation functions and available ESF exists for other Table 15.0-6 events such as inadvertent opening of a SG safety valve, steam line break (SLB), loss of normal feedwater flow, feedwater system pipe break, and uncontrolled reactive rod control cluster assembly (RCCA) bank withdrawal from a subcritical, low power conditions, or at power conditions.

Verify the accuracy of the information provided in Table 15.0-6 and revise the table to be consistent with applicable Sections of Chapter 15.

440.060

List all systems or components that are considered in the transients and accidents analyses for determination of the limiting single failure events and discuss the rationale of selecting the worst single-failure event for each event listed in Table 15.0-7.

440.061

Table 15.0-8 lists the systems or components that are non-safety-related but credited in the accident analysis.

A. Discuss for each of the event category, how these non-safety-related equipment are used for mitigation of transients and what their effects are on determination of the limiting case for each event category presented in Chapter 15.

B. Reference the TS sections for these non-safety-related equipment to address the compliance of Item (c)2(ii)(C) of 10 CFR 50.36 that requires a TS for the systems or components that are used for event mitigation.

440.062

Westinghouse has issued three Nuclear Service Advisory Letters (NSAL), NSAL-02-3 and Revision 1, 02-4, and 02-5, which document the problems with the Westinghouse-designed SG water level setpoint uncertainties. NSAL-02-3 and its revision, issued on February 15 and April 8, 2002, respectively, deal with the uncertainties caused by the mid-deck plate located between the upper and lower taps used for SG level measurements. These uncertainties affect the low-low level trip setpoint (used in the analyses for events such as the feedwater line break, ATWS and steam line break.) NSAL-02-4, issued on February 19, 2002, deals with the uncertainties created because the void contents of the two-phase mixture above the mid-deck plate were not reflected in the calculation and affect the high-high level trip setpoint. NSAL-02-5, issued on February 19, 2002, deals with the initial conditions assumed in the SG water level related safety analyses. The analyses may not be bounding because of velocity head effects or mid-deck plate pressure differential pressure which have resulted in significant increases in the control system uncertainties.

A. Discuss how the AP1000 design accounts for all of these uncertainties documented in these advisory letters in determining the SG water level setpoints.

B. Discuss the effects of the water level uncertainties on the analyses of the LOCA and non-LOCA transients and the ATWS event, and verify that with consideration of all the water level uncertainties, the analyses used to support the design certification are still limiting.

440.063

Section 15.1.2 presents the results of an analysis for the increased feedwater flow event. No figure is presented to show that the calculated DNBRs do not exceed the specific acceptable fuel design limits during the transient.

A. Provide the DNBR figure for the staff to review.

B. No information is presented to address the SG overfill issue. Specifically, for the case initiated from a full opening of a feedwater isolation valve without the isolation valve reclosure because of a single failure consideration, the applicant is requested to identify the safety-related equipment that are credible to isolate the feedwater in order to prevent SG overfill.

C. If non-safety-related systems or components (such as the feedwater control valves or feedwater pumps) are credited to isolate or terminate the feedwater, the applicant should show that the non-safety related system or component is reliable for feedwater isolation and provide a TS LCO to meet the requirements specified in (c)2(ii)(C) of 10 CFR 50.36.

440.064

Section 15.1.3 presents the results of analysis for the excessive steam flow event initiated from rated load. Section 15.1.3 also indicates that two cases are analyzed: one for minimum reactivity feedback and the other for maximum reactivity feedback. Both cases are evaluated with and without automatic rod control.

A. Discuss how the event with the initial power level at rated power bounds the cases initiated from lower power conditions.

B. Provide the values of the moderator temperature and Doppler feedback coefficients assumed in the analysis for the minimum and maximum reactivity feedback, and confirm that the analytical values are bounded by the TS values.

440.065

It states on page 15.1-9 of Chapter 15 that the CMT actuation on an “S” signal is from one of the four signals including low pressurizer level signals. This statement implies that the low pressurizer level signal is an “S” signal. This is inconsistent with Item 1 of TS Table 3.3.2-1, which indicates that “S” signals are from manual initiation, high containment pressure, low pressurizer pressure, low steam line pressure and low RCS cold leg temperature. According to Item 1 of TS Table 3.3.2-1, the low pressurizer level signal is not an “S” signal.

Clarify the inconsistency between the TS and the Chapter 15 analyses related to the definition of an “S” signal for the low pressurizer level signal.

440.066

The statements on pages 15.1-15 and 15.1-22 indicate that a comparison of the results from “the detailed core analysis” with the LOFTRAN predications confirms the overall conservatism of the methodology used for analyses of an SLB event and an inadvertent operation of the PRHR heat exchanger event.

Describe the methods and computer codes used for “the detailed core analysis,” reference the associated NRC acceptance letters to confirm the acceptance of the methods and codes for licensing calculations, and demonstrate that the use of the acceptable methods and codes are within in the applicable ranges.

440.067

It is stated on page 15.1-15 that the limiting SLB case is the complete severance of a steam line, with the plant initially at no-load conditions and full reactor coolant flow with offsite power. However, no quantitative analysis is presented to support the limiting SLB case identified. The guidance for the SLB analysis is provided in SRP 15.1.5. Specifically, Item b of the acceptance criteria states that "Assumptions as to the loss of the offsite power (LOOP) and the time of loss should be made to study their effects on the consequences of the accident. A LOOP may occur simultaneously with the pipe break, or during the accident, or offsite power may not be lost."

Provide analyses of (1) the SLB cases at full power with and without an LOOP and (2) the SLB at no-load conditions with and without an LOOP, and address its compliance of the SRP guidance related to the assumption of the LOOP. The analysis should consider the effects of time of an LOOP (occurred simultaneously with an SLB, or during the transient) on an SLB event and show the calculated DNBRs for both pre-trip and post-trip core conditions at initial power levels of full power and zero power.

440.068

It is stated on page 15.1.15 of Chapter 15 that for the SLB analysis, "the maximum overall fuel-to-coolant heat transfer coefficient is used to maximize the rate of cooldown."

Discuss methods for calculations of overall fuel-to-coolant heat transfer coefficient and demonstrate that the coefficient used in the analysis is a maximum value expected during an SLB event.

440.069

Item d of the acceptance criteria in SRP 15.1.5 states that "the worst single active component failure should be assumed to occur. The assumed single failure may cause more than one SG to blow down, or may be in any of the systems required to control the transient."

Discuss the determination of the worst single active failure and address its compliance with the SRP guidance related to the SLB with steam blowdown from both SGs.

440.070

It states on page 15.1-16 of Chapter 15 that for the SLB analysis, power peaking factor corresponding to one stuck RCCA is determined at the end of core life. The coldest core inlet temperatures are assumed to occur in the sector with the stuck rod.

Discuss the methods used for the power peaking factor determination and address the acceptance of the methods and computer codes used. Provide values of the calculated total power peaking factors and justify that they are conservative for calculations of the minimum DNBRs during an SLB event.

440.071

It states on page 15.1-22 of Chapter 15 that for the analysis of an inadvertent operation of the PRHR heat exchanger, “a negative moderator coefficient corresponding to the end-of-life rodded core” is used.

A. Discuss the methods used for the moderator coefficient determination and address the acceptance of the methods and computer codes used.

B . Provide values of the calculated negative moderator coefficients and associated uncertainties, and address their acceptability for use in the analysis of an inadvertent operation of the PRHR heat exchanger.

440.072

As stated on page 15.2-15 of Chapter 15, in the analysis of the turbine trip event, the initial core power, reactor coolant temperature, and pressure are assumed to be at their nominal values consistent with steady-state full-operation. The use of initial plant parameters without inclusion of measurement uncertainties may not predict a highest peak RCS pressure and may be non-conservative from consideration of the over-pressurization point of view.

Address acceptability of the turbine trip analysis in determination of the peak RCS pressure.

440.073

The analysis of the turbine trip event (Section 15.2.3) identifies that the most limiting DNBR case is the case with a minimum reactivity feedback and without pressurizer spray in combination of an LOOP. However, Section 15.2.3 does not provide calculated DNBRs.

Provide a figure to show the calculated DNBRs for the limiting turbine trip case.

440.074

As stated on page 15.2-11 of Chapter 15, conservative PRHR heat exchanger heat transfer coefficients are used in the analysis for the loss of ac power to the plant auxiliaries event.

A. Discuss the determination of PRHR heat transfer coefficients associated with the low flow rate caused by the RCP trip, and justify the conservatism of the calculated heat transfer coefficients by comparing it with the test data applicable to the temperature, pressure and flow conditions during the loss of an ac power event.

B. Perform a sensitivity study and quantify the effects of measurement uncertainty for heat transfer coefficients on the calculated peak pressure and decay heat removal capability during the transient.

440.075

In consideration of the measurement uncertainty effects, various combinations of initial temperature and pressure are assumed in the analyses. The assumed initial temperatures and pressure are: 7°F below and 50 psi above the normal values for the loss of ac power event (page 15.2-10 of Chapter 15); 7°F and 50 psi below the normal values for the loss of normal feedwater flow (page 15.2-14); and 6.5°F above and 50 psi below the normal values for the feedwater line break event (page 15.2-18). For all three events, the RCS and SG pressures will increase during the events. However, the measurement uncertainties are assumed in different directions (above and below the normal value) for the three events.

Address the acceptability of the initial temperatures and pressures with associated uncertainties assumed in the analyses for these pressurization events.

440.076

The feedwater line break (FLB) event is analyzed for a double-ended rupture of the largest feedwater line, which was previously identified in WCAP-9230,"Report on the Consequences of a Postulated Main Feedline Rupture," as the limiting case, resulting in a highest peak RCS pressure. Considering the plant design differences in the AP1000 and currently operating PWRs, the results of FLB analysis may be different and the limiting FLB case may be different from the one previously identified for currently operating PWRs.

Address the applicability of WCAP-9230 to the AP1000 design for determination of the limiting FLB case, and confirm that the double-ended rupture is the limiting break for the AP1000.

440.077

The guidance for a LOOP assumed for the FLB analysis is provided in SRP 15.2.8. Specifically, Item b of the acceptance criteria states that "Assumptions as to whether offsite power is lost and the time of loss should be make conservatively. Offsite power may be lost simultaneously with the concurrence of the pipe break, the loss may occur during the accident, or offsite power may not be lost."

Discuss the determination of the time of an LOOP assumed for the limiting FLB analysis and address the compliance with the SRP guidance related to the time of an LOOP.

440.078

The semi-scale test data (Section 4.3.3.1 of NUREG/CR-4945) shows that the SG heat transfer capacity remains unchanged until the SG liquid inventory is nearly depleted. This is followed by a rapid reduction to zero percent heat transfer with little further reduction in the SG liquid inventory.

Discuss the SG heat transfer model used in the FLB analysis and verify that the heat transfer model is conservative as compared to the semi-scale test data.

440.079

Page 15.2-26 of Chapter 15 indicates that for the loss of ac power event, the CMT actuates on the low RCS cold leg temperature (T_{cold}) “S” signal at 4753 seconds followed by the closure of the steam line isolation 12 seconds later at 4765 seconds. Page 15.2-27 shows that for the loss of normal feedwater flow event, the steam line isolation occurs on the low T_{cold} “S” signal at 1160.6 seconds followed by the CMT actuation 11 seconds later at 1171.6 seconds. Even though both the CMT and steam line isolation are actuated on low T_{cold} “S” signals, the sequence of the CMT and steam line isolation actuations and the delay time between them are different for the loss of ac power and loss of normal feedwater flow events.

Provide reasons for the differences in the event sequences and delay times.

440.080

Section 15.3.3 presents the results of analysis for the locked rotor event. The analysis assumes that DNB starts to occur at the beginning of the accident. In accordance with the SRP 4.4 guidance, all fuel rods experiencing DNB should be assumed to fail.

Calculate the number of failed fuel rods in accordance with the SRP 4.4 guidance and verify that the calculated failed rod number is within the assumed value used for the radiological calculations.

440.081

Section 15.4.3.2.2.2 indicates that the calculated minimum DNBR for the withdrawal of single full-length RCCA event is less than the safety limit. As a result, the fuel rods predicted to fail is less than 5 percent of the total fuel rods.

A. Provide a figure showing the calculated DNBRs during the transient.

B. Discuss the analytical methods, input parameters (such as pin census data and peak factors) and assumptions used to determine the number of failed fuel rods, and show that the methods used for the analysis are acceptable and the input parameters and assumptions are conservative with respect to the fuel failure calculations.

440.082

Section 15.4.3.3 indicates that for cases of dropped RCCAs, dropped RCCA bank, and statically misaligned RCCA, the calculated DNBRs remain greater than the safety DNBR limits. However, no calculated DNBRs are presented.

Provide figures showing calculated DNBRs during the transients and demonstrate that no fuel rod is predicted to fail for the events of RCCA drop and RCCA misalignment.

440.083

As indicated in Section 15.4.6, minimum reactor coolant water volumes are used in the boron dilution event analysis. The values of the RCS water volumes are 8126 cubic feet (ft^3) for Modes 1 and 2; 7300 ft^3 for Mode 3; 2805 ft^3 for Mode 4 and 2402 ft^3 for Mode 5.

Specify the water volumes in the reactor vessel, SGs and RCS pipes used to calculate the minimum RCS water volumes and justify that the RCS water volumes used in the analysis for each mode are conservative and acceptable.

440.084

Page 15.4-31 of Chapter 15 indicates that the system over-pressure analysis for the rod ejection event is performed with "plant transient computer code."

Reference the associated NRC acceptance letters and confirm the acceptance of the "plant transient computer code" for licensing calculations.

440.085

Page 15.5-3 of Chapter 15 indicates that in the over-pressure analysis for the CMT inadvertent operation event, "PRHR heat transfer capacity has been minimized."

Describe the model that is used to calculate a minimized PRHR heat transfer capacity and demonstrate its conservatism by comparing with heat transfer test data applicable to the AP1000 PRHR heat exchanger.

440.086

As indicated on pages 15.5-3 and 15.5-7 of Chapter 15, the assumed initial temperatures and pressure are: 7°F and 50 psi below the normal values for the CMT malfunction event, and 6.5°F and 50 psi above the normal values for the CVS malfunction event. Both events are analyzed to predict overpressure during the transients. The assumed initial temperatures and pressures show that measurement uncertainties are assumed in different directions (below and above the normal values) for these events.

Discuss the criteria used to select initial temperatures and pressures for the analyses and justify that the temperatures and pressures so selected are conservative and acceptable.

440.087

Section 15.6.2 indicates that the break flow rates are limited to 100 gallons-per-minute (gpm) and 130 gpm for the CVS charging line break and the RCS sample line break, respectively.

Discuss the models used to calculate these break flow rates and show that the break flow rates are overpredicted when the temperature and pressure conditions and break sizes are considered in the break flow rate calculation.

440.088

Page 15.6-10 of Chapter 15 indicates that the modified LOFTT2 code described in WCAP14234 is used for the steam generator tube rupture (SGTR) analysis.

Reference the associated NRC acceptance letters and confirm the acceptance of the modified LOFTTR2 code for the AP1000 licensing calculations. Also, verify that use of the computer code for the SGTR analysis is within the applicable range of the NRC-approved code.

440.089

Page 15.6-11 of Chapter 15 indicates that in the SGTR analysis, the ruptured SG power-operated relief valve (PORV) is assumed to fail open when the low-2 pressurizer level "S" signal generates, while page 15.6-12 and Table 15.6.3-1 shows that the failure of the PORV occurs on the low pressurizer pressure "S" signal.

Provide the rationale for selection of the time of the SG PORV to fail and show that the selected PORV failure time results in a maximum RCS mass release and is conservative for the SGTR analysis. Correct any inconsistencies in Chapter 15 for the PORV failure time.

440.090

According to the description on Section 7.3.1.2.4, the ADS-4 consists of four parallel paths. The four paths are divided into two redundant groups with two paths in each group. Within each group, one path is designed to be Substage A and the second path is designed to be Substage B. Therefore, there are two paths for each of Stage-4A and Stage-4B ADS. Table 15.6.5-7, "AP1000 ADS Parameters," indicates that the number of paths for Stage-4A ADS is one out of two.

Explain the inconsistency in Table 15.6.5-7 for the number of paths for Stage-4A ADS.

440.091

In the discussion and analysis of the double-ended direct vessel injection (DEDVI) line break (Section 15.6.5.4C.2), it was assumed that the ADS-4A valve failed to open (single failure) and that the containment pressure is at the WGOTHIC calculated minimum. These conditions may be conservative for depressurization but not from the point of view of long-term cooling. Consider the case when all ADS-4 valves open, with a maximum containment pressure. Steam velocity in the ADS-4s will be minimum.

Will that steam velocity be able to entrain and remove liquid from the core? (Note, it is not feasible to draw this conclusion from the information in the code applicability report without extensive calculations).

440.092

In the case of the DEDVI break and wall-to-wall floodup (Section 15.6.5.4C.3), it was estimated that 28.5 days will be required to attain this condition.

How was this time estimated? How was the in leak rate derived? Would the long-term cooling be sustainable if the floodup was assumed to occur at the end of the IRWST injection?

440.093

Please include the units in Table 15.6.5-1

440.094

Tables 15.6.5-10 and -11. Accumulator injection start; is it out of sequence or is there a misprint in the initiation time?

440.095

It is not evident that the choice of the DEDVI for the demonstration of long-term cooling is the most conservative case. The case of a direct vessel injection (DVI) break achieves IRWST injection early with relatively high decay heat. However, a great deal of water is injected through the DVI break after the sump water level achieves the break elevation.

If the transient did not involve a DVI break, would there be sufficient water to keep the core covered through the IRWST injection?

440.096

Throughout the discussion of long-term cooling, there are water levels indicated but no mention is made of a reference point. For example, in Section 15.6.5.4C.3, a level of 103.5 ft is given with no reference point. Likewise, in the Figures (from 15.6.1-1 to 15.6.5.4C-28) there are water levels but no reference points are identified, making it difficult or impossible to conclude if the core is covered in the time segment indicated.

Please correct this deficiency.

440.097

The documentation of the large break LOCA (LBLOCA) analysis methodology and results in Section 15.6.5.4A is totally inadequate.

Please include additional information comparable in content and detail to the small-break LOCA (SBLOCA) and the long-term cooling.

440.098

SB LOCA analyses are performed for inadvertent ADS actuation, a 2-inch break in a cold leg, the double-ended rupture of a DVI line, and a 10-inch cold leg break. 10 CFR 50.46 requires analysis of a number of postulated LOCA of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated LOCA are calculated. Analyses described in Section 7 of WCAP-14869, “MAAP4/NOTRUMP Bench marking to Support Use of MAAP4 for AP600 PRA Success Criteria Analyses,” demonstrate that for the AP600, small-breaks in the hot leg lead to lower levels of reactor vessel inventory than do cold leg break locations. This is because for the AP600, the cold legs are at a higher elevation than the hot legs. The AP1000 design has the same coolant loop arrangement as the AP600 design. Please provide SB LOCA analyses for a complete spectrum of hot and cold leg break sizes to ensure that the most severe SBLOCAs are calculated. Please include postulated breaks in the CMT pressure balance lines.

440.099

During review of the AP600, the NRC staff raised the issue of boron dilution associated with the SB LOCA reflux condensation, the so called “Finnish Scenario.” The staff requested Westinghouse to address the issue in a letter to Westinghouse entitled “AP600 Boron Dilution Transient Analyses,” dated September 2, 1996 (AP600 request for additional information (RAI) No. 440.120). The staff is again requesting Westinghouse to address the same issue as it applies to the AP1000. Westinghouse should also address subsequent concerns that were raised by the staff as a consequence of the incomplete response to the September 2, 1996, letter. The subsequent letters referred to here were dated May 14, 1997, “Revised Response to RAI 440.120 for Rapid Boron Dilution Scenarios” October 1, 1997; “Response to Request for Additional Information on Mixing in Downcomer and Lower Plenum” (RAI 440.724); and January 16, 1998, “AP600 Response to FSER [final safety evaluation report] Open Items.”

440.100

AP1000 TS Surveillance Requirement (SR) 3.1.4.3 requires verification of each control rod drop time, from the fully withdrawn position, to be $\leq$ [2.7] seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with $T_{avg} \geq 500°F$, and all RCPs operating. Though the rod drop time of 2.7 seconds is bracketed, indicating that it is preliminary value to be replaced by the COL applicants with final plant specific value, this value is inconsistent with the value of 2.47 seconds assumed in Chapter 15 design-basis transients and accidents analyses as shown in Figure 15.0.5-1.

Explain the difference in the rod drop times of SR 3.1.4.3 and Figure 15.0.5-1.

440.101

In TS 3.2.1, “Heat Flux Hot Channel Factor (F_Q (Z)) (F_Q Methodology),” the Required Actions for Conditions A and B are different from those specified in TS 3.2.1B, “Heat Flux Hot Channel Factor (F_Q (Z)) (RAOC - W(Z) Methodology),” of NUREG-1431, Rev. 2, “Standard Technical Specifications, Westinghouse Plants” (WSTS). For example, WSTS Required Action A.4 requires the performance of SR 3.2.1.1 and 3.2.1.2, whereas AP1000 TS Required Action A.4 requires the performance of SR 3.2.1.1 only. Also WSTS Required Actions B.2, B.3, and B.4 are not included in the AP1000 TS.

Explain the differences and why they are acceptable for the AP1000.

440.102

TS LCO 3.2.5 specifies the operating limits of the power distribution parameters (peak kw/ft, $F_{\Delta H}^{N}$, and DNBR) monitored by the On-line Power Distribution Monitoring System (OPDMS). TS 5.6.5 lists WCAP-12472-P-A, “BEACON - Core Monitoring and Operation Support System,” August 1994, and Addendum 1, May 1996, as the approved method used for the determination of the monitored power distribution parameters limits. TS 5.6.5 also contains a “REVIEWER’S NOTE” stating that “additional power distribution control and surveillance methodologies (for MSHIM and OPDMS monitoring) are currently under development and will be added upon NRC approval....” Section 4.3.4 of Design Control Document (DCD) states that the COL applicant will reference an NRC-approved addendum to WCAP-12472-P-A covering AP1000 fixed incore detector.

Though the BEACON system described in WCAP-12472-P-A has been accepted by NRC for performing continuous on-line core monitoring and operations support functions for Westinghouse PWRs, its acceptance is limited to the current standard Westinghouse OPDMS with the use of movable incore detectors, on which the instrumentation data base in WCAP-12472-P-A and the staff evaluation were based. Since the AP1000 OPDMS uses fixed in-core detectors, in-core thermocouples, and loop temperature measurements, which differ sufficiently from these data base, an evaluation is required for the generic uncertainty components to determine if the assumptions made in the BEACON uncertainty analysis remains valid, and assure that the power peaking uncertainties for the enthalpy rise and heat flux provide 95 percent probability upper tolerance limits at the 95 percent confidence level. (Section 4.3.2.2.7 discusses experimental verification of power distribution analysis.)

When will Westinghouse submit the addendum to WCAP-12472 on the AP1000 fixed incore detector?

440.103

In TS Table 3.3.1-1, “Reactor Trip System Instrumentation,” and Table 3.3.2-1, “Engineered Safeguards Actuation System Instrumentation,” the “Trip Setpoint” values are [bracketed], indicating they are preliminary reference values to be replaced by the COL applicants with final plant specific values. The majority of the reference values in these tables are the same as limiting setpoints assumed in the design-basis accident analyses shown in DCD Table 15.0-4a, “Protection and Safety Monitoring System Setpoints and Time Delay Assumed in Accident Analyses.” Some are quite different from Table 15.0-4a. For example, the reactor trip setpoints on low SG narrow range level and High-2 SG level are 95,000 pounds mass (lbm) and 100 percent, respectively, in Table 15.0-4a, and 45000 lbm and 95 percent, respectively, in Table 3.3.1-1.

Westinghouse Setpoint Methodology for Protection defines the TS nominal trip setpoint (NTS) as the Safety Analysis Limits (SAL) adjusted for (plus or minus, depending on which way is more restrictive) the total allowance (TA) for the instrumentation uncertainties.

A. Use of the SAL as the preliminary reference values of the trip setpoint could lead to potential mistakes by the COL applicant to use the SAL as the plant-specific NTS values. To avoid the potential for mistakes, a note should be added to TS Tables 3.3.1-1 and 3.3.2-1 to clearly describe how the NTS values are determined.

B. For all of the reactor trip setpoints and the engineered safety feature actuation system (ESFAS) trip setpoints in Tables 3.3.1-1 and 3.3.2-1, respectively, provide safety analysis limit value, the total allowance for instrumentation uncertainties, and the nominal trip setpoint in the TS.

C. Explain the reason why the “Allowable Value” columns are left blank in Tables 3.3.1-1 and 3.3.2-1.

D. To assist the COL applicant in the determination of the allowable values, a note should be added to these tables to define the allowable values as adding (or subtracting) the calibration accuracy of the device tested during the channel operational test to the NTS in the non-conservative direction for the application.

440.104

Several TSs contain limiting values that are inconsistently denoted by the values within a bracket ([ ]), indicating they are preliminary, not the final plant specific values. For example, in TS 3.4.1, “RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits,” brackets are used in LCO limiting values of pressurizer pressure, RCS average temperature, and RCS total flow rate, and in SR 3.4.1.4 on the RCS total flow surveillance value, but not for the surveillance values of the pressurizer pressure, RCS average temperature, and RCS total flow specified in SRs 3.4.1.1, 3.4.1.2, and 3.4.1.3, respectively. In TS 3.4.2, “RCS Minimum Temperature for Criticality,” a bracket is included in SR 3.4.2.1 surveillance limit of the RCS average temperature, but not for the LCO limit value.

Explain the inconsistency and make corrections if necessary.

440.105

TS Bases B 3.4.1 LCO section describes a measurement error contained in the RCS total flow rate based on performing a precision heat balance and using the result to normalize the RCS flow indicator.

Explain why there is no discussion in the BASES of the bias that could arise from potential fouling of the feedwater venturi.

440.106

TS LCO 3.4.9 specifies that at least one RCP shall be in operation with a total flow through the core of at least 10,000 gpm while in MODES 3, 4 and 5, whenever the reactor trip breakers are open. SR 3.4.9.1 requires verification that at least one RCP is in operation at ≥25 percent rated speed or equivalent. TS BASES 3.4.9 provides a table of pump percentage rated speeds as a function of number of pumps operating that will deliver the required minimum flow.

The minimum RCS flow limit is an initial condition in the design-basis analysis of a possible boron dilution event to provide a mixing of the inadvertent diluted water with the primary flow. In the safety analysis of boron dilution events during MODES 3, 4, or 5, operation, Section 15.4.6.2 states that the RCS dilution volume is considered well-mixed. The TSs require that when in MODES 3, 4, 5, at least one RCP shall be operable, which provides sufficient flow through the system to maintain the system well-mixed. As shown in Table 5.4-1, the AP1000 RCP design flow is 78,750 gpm per pump.

A. Provide analysis or test data to demonstrate that the 10000 gpm minimum mixing flow specified in LCO 3.4.9 is sufficient to provide a well-mixed flow condition in the boron dilution events to validate the safety analysis assumptions.

B. Provide the characteristics or specification of the variable speed pump design that ensure the minimum mixing flow will be delivered with the pump percentage rated speeds shown in the TS BASES.

440.107

TS 5.6.5 lists WCAP-14807, “NOTRUMP Final Validation for AP600,” and WCAP-15644, “AP1000 Code Applicability Report,” as the approved analytical methods used to determine the heat flux hot channel factor. However, the NRC review of WCAP-15644 has identified possible deficiencies for the AP1000 application in the NOTRUMP entrainment models at the time of ADS-4 actuation, as described in a letter from James Lyons to W. E. Cummins, “Applicability of AP600 Standard Plant Design Analysis Codes, Test Program and Exemptions to the AP1000 Standard Plant Design,” dated March 25, 2002.

Appropriate approved reports should be listed when the final resolution of the application of the AP600 codes to the AP1000 design is reached.

440.108

SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs,” dated March 28, 1994, discusses the process for the resolution of regulatory treatment of non-safety systems (RTNSS), including the methodology for the selection of important non-safety systems, their reliability/availability missions, and regulatory oversight. Section 16.3 of DCD, “Investment Protection,” specifies the investment protection short-term availability controls for the important non-safety systems identified through the RTNSS process. Section 16.3.1 states that the importance of non-safety-related systems, structures and components (SSCs) in the AP1000 has been evaluated, using probabilistic risk assessment (PRA) insights to identify structures, systems, and components (SSCs) that are important in protecting the utilities investment and for preventing and mitigating severe accidents.

Provide a detailed AP1000-specific evaluation in accordance with the RTNSS process described in SECY-94-084 for identification of the important non-safety systems and their reliability/availability missions, which forms the basis for Section 16-3 investment protection short-term availability controls.

The following questions are related to Tier 2 Information, Appendix 19E, “Shutdown Evaluation.”

440.109

Section 18.9 states that WCAP-14690, “Designer’s Input To Procedure Development for the AP600,” Revision 1, June 1997, provides input to the COL applicant for the development of plant operating procedures, including information on the development and design of the AP600 emergency response guidelines (ERGs) and emergency operating procedures (EOPs). It applies directly to the AP1000 since the AP1000 is operated in the same manner as the AP600. Sections 19E.1.2 and 19E.3.3 reference the AP600 ERGs to address the shutdown operation issues for the AP1000 design, and states that the AP600 ERGs are applicable to the AP1000 for the purpose of developing EOPs.

A. Provide a discussion on the effects of the significant design changes from the AP600 to the AP1000 on the applicability of the AP600 ERGs to the AP1000 design. Provide technical bases to justify that the AP600 ERGs are applicable to the AP1000 emergency response procedures, including shutdown operation issues.

B. Discuss the applicability of the transient data presented in background sections of AP600 ERGs to the AP1000 design.

C. Describe the verification and validation (V&V) program for the AP1000 ERGs.

440.110

Section 19E.2.1.2.6 indicates that the design pressure of 40 psia for the SG nozzle dam is selected to withstand the RCS pressures that can occur during a loss of shutdown cooling event.

Discuss the analysis of the loss of shutdown cooling event to show that the calculated peak RCS pressure is within the design pressure of 40 psia for the nozzle dam.

440.111

As stated in Section 19E.2.2.2.2, the AP1000 safety-related actuations include the signal to isolate the main steam line on a high negative rate of change in steam pressure. This safety-related signal is provided to address a steam line break (SLB) that could occur in Mode 3 or 4. Item 2.c(2) of TS Table 3.3.2-1 specifies that the steam line isolation signal (SLIS) on a high negative steam line pressure is required to be operable for only Mode 3 conditions.

Explain why the applicability of the TS requirements for the SLIS does not include Mode 4 conditions.

440.112

As indicated on page 19E-10, for non-LOCA transients, the passive core cooling system (PXS) in conjunction with the passive containment cooling system (PCS) has the capability to establish safe shutdown conditions, cooling the RCS to less than 420°F within 36 hours, with or without the RCPs operating. The staff finds that none of the analyses for the non-LOCA events have demonstrated that the RCS is cooled down to less than 420°F (see results on pages 15.2-58, -58, -69,-70, -79 and -80 etc.)

Provide a discussion of the PXS analysis to show that the PXS can be used to cool down the RCS to less than 420°F within 36 hours as designed for non-LOCA transients.

440.113

As indicated in Section 19E.2.4.2.1, a large net positive suction head (NPSH) provides the RNS pump capacity during mid-loop operations with saturated fluid in the RCS without throttling the RNS flow.

Discuss the required NPSH for this configuration.

440.114

The analysis of the loss of RNS during mid-loop operation shows (in Table 19E.2-1) that the available operator time for the core uncovery prevention is 40 minutes. The applicant claims that the operators have a sufficient amount of time to actuate gravity injection before core uncovery.

Provide a discussion of the required operator actions, and show that clear indications as well as appropriate operating procedures are available; and that the operator has a sufficient time to prevent the core from uncovering for a loss of RNS during mid-loop operation.

440.115

As stated on page 19E-24, the spurious opening of a SG safety valve is considered as a credible SLB during low power and shutdown conditions, and is analyzed for the AP1000 design certification. Any SLB events with break sizes greater than the SG safety valve are considered as non-credible SLBs and are not analyzed.

Provide technical bases to justify that larger break SLBs (such as SRP 15.1.5 events) are not credible SLB events during the low power and shutdown conditions.

440.116

As indicated on page 19E-27, in Modes 2 through 4, the transient response to a loss of condenser vacuum or inadvertent main steam isolation valve (MSIV) closure is bounded by the turbine trip analysis from full power because the power mismatch is low.

Explain why the power mismatch is lower for a loss of condenser vacuum or inadvertent MSIV closure as compared to the turbine trip event at full power conditions.

440.117

It states in Section 19E.4.3.3 that the loss of normal feedwater event initiated from full power conditions is mitigated by tripping the reactor on low steam generator level (LSGL). Following reactor trip, the PRHR HX is actuated on LSGL for heat removal. The staff notes that the LSGL (narrow range) setpoint (Item 13 in TS Table 3.3.1-1) is 45000 lbm of water in the SG, and the LSGL (wide range) setpoint to actuate the PRHR (Item 13.c of TS Table 3.3.2-1) is 55,000 lbm of the water in the SG. Based on the TS setpoints, it appears that the PRHR will actuate before the reactor trips on the LSGL.

Clarify the inconsistency of the sequence for occurrence of the reactor trip and actuation of the PRHR in the TS and Section 19E.4.3.3. (This question is also applied to Section 19E4.3.4, Feedwater System Pipe Break.)

440.118

As stated on page 19E-33, WCAP-10698-A, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," concludes that for standard Westinghouse PWRs, zero power and low mode SGTR overfill analyses are not limiting, based on more rapid operator responses expected in those conditions. It further states that when operator actions are credited for AP1000 SGTR mitigation, the plant behaves in a manner comparable to a standard Westinghouse PWR and the conclusions of WCAP-10698-A apply to the AP1000.

Discuss a comparison of applicable analyses to demonstrate that the AP1000 plant behaves in a manner comparable to a standard Westinghouse PWR.

440.119

It indicates on pages 19E-34 and-35 that Reference 10 ("AP600 Shutdown Evaluation Report") of Appendix 19E documents analyses of LOCA events and loss of RNR events at lower modes for the AP600. It also indicates that Reference 10 for the AP600 is applicable to the AP1000 because (1) accident analyses presented in Chapter 15 demonstrated that the AP1000 plant response to accidents is similar to the AP600 plant response, and (2) availability of the passive core cooling system components in lower modes is the same for both the AP600 and the AP1000.

Discuss a comparison of applicable Chapter 15 analyses to demonstrate that the AP1000 plant response to accidents is similar to the AP600 plant response. In lower modes, the AP1000 plant response to accidents may be different from the AP600 plant response. Explain why the use of Chapter 15 accident analyses (which are performed for Modes 1 and 2 conditions) is acceptable for justifying the applicability of the cited reference to the AP1000 plant at lower modes.

440.120

On page 19E-3, it indicates that two RCS hot-leg level channels are available to monitor the RCS water level during mid-loop operation.

Discuss the measurement uncertainties of the hot-leg level system and confirm that they are adequately included in the low level setpoints for isolation of letdown flow and actuation of IRWST injection and fourth-stage ADS valves. Reference the TS that includes those setpoints.

440.121

Page 19E-4 documents information related to RCS temperature detectors for shutdown conditions.

Describe how these detectors are used during shutdown, mid-loop, and accident conditions.

440.122

As indicated on page 19E-5, the applicant relies on the test data and analysis (Reference 4 of Section 19E.1) performed for the AP600 to show the adequacy of the AP1000 step-nozzle design to minimize vortex formation and air entrainments into the pump suction. Justify that the cited reference is applicable to the AP1000 design.

440.123

Section 6.7 of NUREG-1449, “Shutdown and Low Power Operation at Commercial Nuclear Plant in the United States,” describes instances in which the failure of temporary RCS boundaries (such as freeze seal, which is used to temporarily isolate fluid systems and temporary plugs for neutron instrument housing) can lead to a rapid non-isolable loss of reactor coolant.

Address this concern with respect to failure of temporary boundaries in the AP1000.

440.124

Current plants use temporary reactor cavity seals to flood the refueling cavities. Failure of these seals can divert water to the reactor pit, and subsequently to the reactor drains, and may result in a loss of shielding and fuel cooling during spent fuel assembly movement.

Address the ability to quickly move and safely store fuel assemblies during a seal failure event.

440.125

In NUREG/CR-5820, “Consequences of the Loss of the Residual Heat Removal System in Pressurized Water Reactors,” NRC describes a loss of residual heat removal event that could lead to the core uncovery because of a lack of coolant circulation flow. Such conditions could occur during the flooding of the refueling pool cavity while preparing for fuel shuffling operations. Under these conditions, the vessel upper internals may provide sufficient hydraulic resistance to natural circulation between the refueling pool and the reactor, and may prevent the refueling water from cooling the core if the residual heat removal (RHR) cooling is lost.

Address this NUREG/CR-5820 issue and show the AP1000 design is adequate to preclude pressurization of the RCS in Mode 6 following a loss of the RNS event.

440.126

NRC Information Notice (IN) 92-54, “Level Instrumentation Inaccuracies Caused by Rapid Depressurization,” describes the issue related to potential problems of noncondensible gases in hot-leg level instrument lines. The applicant indicates on page 19E-3 that the IN 92-54 issue has been addressed in the layout of the instrument lines.

Identify the features in the level instrument lines that are designed to address the IN 92-54 issue and show the issue is satisfactorily resolved.

440.127

NUREG-0737, “Clarification of TMI Action Plan Requirements,” TMI Action Item II.F.2 requires that instruments be provided that provide in the control room an unambiguous indication of inadequate core cooling, such as primary coolant saturation meters in PWRs, and a suitable combination of signals from indicators of coolant level in the reactor vessel and in-core thermal couples.

Since the AP1000 design does not have a reactor vessel level indication system (RVLIS) as do the current Westinghouse PWRs, provide a detailed discussion, in addition to the brief AP1000 response described in Tier 2 Section 1.9.3 (2)(xviii), on how the AP1000 design conforms to this requirement.

440.128

NUREG-0933, “A Prioritization of Generic Safety Issues,” Task Action Plan Item USI A-17 addresses the concerns of adverse systems interactions (ASI) among various structures, systems, and components (SSCs) in a plant, and identifies the need to investigate the possibility that unrecognized subtle dependencies among the SSCs have remained hidden and could lead to safety significant events. The staff concluded that occurrence of an actual ASI or the existence of a potential ASI, as well as the potential overall safety impact, are very much a function of an individual plant’s design and operational features. Therefore, for new plant designs with new or different configured passive and active systems, such as the AP600 and the AP1000 designs, the staff believes the designer should perform a systematic search for ASIs, and propose resolutions for any that are discovered. For the AP600 design, Westinghouse submitted topical report WCAP-14477, Revision 1, “The AP600 Adverse System Interaction Evaluation Report,” to identify possible adverse interactions among safety-related systems and between safety-related and non-safety-related systems, and to evaluate the potential consequences of such interactions.

Provide a systematic evaluation of the ASI for the AP1000 design, similar to WCAP-14477, Revision 1, or provide detailed justifications, considering the differences between the AP600 and the AP1000, on why the ASI evaluation performed for the AP600 design and conclusion are applicable to the AP1000 design.

<u>The following questions are related to WCAP-15833, “WCOBRA/TRAC AP1000 ADS-4/IRWST Phase Modeling”</u>

440.129

In WCAP-15833, Section 2.2.2, “Separate Effects Verification,” describes assessment of the horizontal stratified flow model in WCOBRA/TRAC using data from the Lim test facility. Upper plenum/hot leg entrainment during the post-ADS period is given a high importance in the phenomena identification and ranking table (PIRT). The maximum steam velocity in the Lim tests is stated to be 18 meters-per-second (m/s) and the maximum water velocity is stated to be 41 centimeters-per-second (cm/s). NRC staff calculations for the AP1000 using RELAP5 predict hot leg velocities for both steam and water that are in excess of those in the Lim tests for the period following ADS-4 actuation during a SB LOCA. Please justify that the steam and water velocities in the hot legs of the AP1000 following SB LOCA will be within the experimental range of the Lim tests or provide other justification for the horizontal stratified flow model in WCOBRA/TRAC.

440.130

Figure 2-33 provides the WCOBRA/TRAC prediction of upper plenum collapsed liquid level for Oregon State University (OSU) test SB18. Comparisons with test data are not given. The applicable test data is given in Figure 5.1.2-12 of WCAP-14252, “AP600 Low-Pressure Integral Systems Test at Oregon State University Final Data Report.” The values of upper plenum liquid level for the test are considerably lower than the WCOBRA/TRAC predictions. Discuss the cause of this apparent non-conservatism of WCOBRA/TRAC and its implications on SB LOCA analysis for the AP1000.

440.131

Figure 5.1.2-4 of WCAP-14252 provides upper plenum void fractions for OSU test SB18. Please provide the corresponding predictions of upper plenum void fractions from WCOBRA/TRAC.

440.132

Figure 5.1.2-37 of WCAP-14252 provides IRWST injection rates for OSU test SB18. Please provide the corresponding predictions for IRWST injection initiation times and flow rate from WCOBRA/TRAC. Were IRWST flow rates predicted by the code or input from the test data?

440.133

Section 3.1.13 states that for the AP1000 analysis, CMT and accumulator injection to the reactor vessel are input to WCOBRA/TRAC using a FILL component. Please describe how these values are obtained. Include all equations and experimental justification for any standalone calculations.

440.134

Section 3.1.14 states that ADS 1-4 and broken pipe boundaries are described using WCOBRA/TRAC BREAK components. The WCOBRA/TRAC Users Manual is referenced. The Users Manual describes BREAK components as a pressure boundary condition and provides a number of user options. Please describe and justify all the options used. Provide a comparison of the methodology used by WCOBRA/TRAC to that used by NOTRUMP. Provide rationale for the greater ADS-4 depressurization calculated by WCOBRA/TRAC in comparison to NOTRUMP. See Figures 3-9 and 3-18.

440.135

Section 3.1.15 states that the initial conditions used in the WCOBRA/TRAC simulations of the AP1000 are the NOTRUMP values at the time of ADS-4 initiation. Figures 3-16 and 3-25 show that WCOBRA/TRAC was initialized at larger values of reactor vessel mass than predicted by NOTRUMP. Please discuss the reasons for this discrepancy.

The following questions are related to AP1000 Tier 1 Information:

440.136

Section 1.3, “Figure Legend,” of Tier 1 Information provides a list of conventions used in the Tier 1 figures, which are somewhat different from the conventions described in Section 1.7 of Tier 2 Information for the Tier 2 Piping and Instrumentation Diagrams (P&ID). Also, some Tier 1 figures actually use Tier 2 conventions which are not defined in Tier 1 Section 1.3. For example, the conventions for the air-operated valve or pneumatic operator are different between Tier 1 and Tier 2; and, in Figure 2.3.6-1, “Normal Residual Heat Removal System (RNS),” the RNS pump mini-flow air-operated isolation valves, RNS-PL-V057A and -V057B, use the Tier 2 convention, which is not defined in Tier 1.

Explain why it is necessary to use different conventions for Tier 1 and Tier 2 information, respectively, and make corrections to the Tier 1 figures to ensure they are consistent with the Tier 2 conventions.

440.137

Many figures in Tier 1 Information show valves status inconsistent with the normal positions of the design arrangement shown in the Tier 2 P&IDs. For example, in Tier 1 Figure 2.1.2-1, “Reactor Coolant System,” the pressurizer safety valves, the pressurizer spray valves, the reactor vessel heat vent valves, the ADS stages 1, 2, and 3 isolation valves and depressurization control valves, and the ADS-4 squib valves, which are all normally closed, are indicated as open. In Figure 2.2.3-1, “Passive Core Cooling System,” the air-operated valves on the PRHR outlet line and the CMT injection lines, and the squib valves on the ADS stage 4 discharge lines, the IRWST injection lines, and containment recirculation lines, which are all normally closed, are shown as open.

Explain why the Tier 1 Information figures do not show the valves’ normal positions and make revisions as necessary to be consistent with the Tier 2 P&IDs.

440.138

For the design evaluation of the RCS pressure relief devices, Tier 2 Information, Section 5.4.9.3, states that in certain design-basis events, where the RCS pressure is slowly increasing as a result of the mismatch between the decay heat generation and removal rates, the pressurizer safety valves are predicted to operate with very low steam flow rates; operation of safety valves under these conditions could result in small leakage from the valve (much less than the normal makeup system capacity).

Explain why Tier 1 Information Section 2.1.2, “Reactor Coolant System,” does not include the pressurizer safety valve design basis of low valve leakage rate as a result of operation near the valve set pressure in the design description and inspection, test, analyses, and acceptance criteria (ITAAC) design commitment.

440.139

The reactor vessel head vent system (RVHVS) valves described in Tier 2 Section 5.4.12 are used to remove noncondensible gases or steam from the reactor vessel head to mitigate a possible condition of inadequate core cooling or impaired natural circulation through the SGs resulting from the accumulation of noncondensible gases in the RCS. The design of the RVHVS is in accordance with the requirements of 10 CFR 50.34(f)(2)(vi).

Explain why there is no design description in Tier 1 Information, Section 2.1.2, and design commitment in ITAAC Table 2.1.2-4 regarding the RVHVS valves.

440.140

In ITAAC Table 2.1.2-4, Item 8d, the acceptance criteria for the ADS design are as follows:

- The calculated ADS piping flow resistance from the pressurizer through the sparger with all ADS Stages 1-3 valves of each group open is $\leq$ 2.92E-6 ft/gpm^2, and the calculated flow resistance for each group of Stage 4 valves and piping is $\leq$ 1.71E-7 ft/gpm^2.
- The effective flow areas through Stages 1, 2, 3, and 4 valves are $\geq$ 4.6 square inches (in^2), $\geq$ 21 in,^{2} $\geq$21 in,^{2} and $\geq$67 in,^{2} respectively.

A. Describe how the ADS piping flow resistance acceptance criteria are determined and explain if they are consistent with the assumptions in the design basis accident analyses.

B. Describe how the ADS effective flow areas acceptance criteria are determined, and explain if they are consistent with the ADS valve design of 4," 8," 8," and 14," for Stages 1, 2, 3, and 4 valves, respectively, as well as consistent with the safety analysis input.

440.141

ITAAC Table 2.1.2-4, Item 9.a shows the acceptance criteria for the RCS flow as “the calculated post-fuel load RCS flow rate $\geq$ 296,000 gpm,” with the flow measurement uncertainties accounted for. As shown in Tier 2 Table 5.1-3, 296,000 gpm is the RCS thermal design flow with 10 percent SG tube plugging. Without SG tube plugging, the RCS thermal design flow is 299,880 gpm.

Provide a clarification in the ITAAC RCS flow rate acceptance criteria with respect to the SG tube plugging condition.

440.142

The AP1000 reactor core consists of 157 fuel assemblies (FA), 53 rod control cluster assemblies (RCCA), 16 gray rod cluster assemblies (GRCA) which are used in load follow maneuvering, and 69 control rod drive mechanisms (CRDM). Tier 1 Information Section 2.1.3 provides the design description and the Design Commitment (in Table 2.1.3-2) of the FAs, RCCAs and CRDMs. Tier 1 Tables 2.1.3-1 and 2.1.3-3 list the tag numbers, American Society of Mechanical Engineers (ASME) Code Section III classification, seismic categories, and locations of the FAs, RCCAs, and CRDMs.

A. Explain why the GRCAs are not included in the Tier 1 Table 2.1.3-2 design commitment to be designed and constructed in accordance with the principal design requirements.

B. Explain why the tag numbers and locations for the GRCAs are not included in Tier 1 Tables 2.1.3-1 and 2.1.3-3.

440.143

Tier 1 ITAAC Table 2.2.3-4, Item 8.c, states that a low-pressure injection test and analysis for each CMT, accumulator, IRWST injection line, and containment recirculation line, as well as the CMT cold leg balance line, will be conducted. The acceptance criteria for each injection line or pressure balance line flow resistance (R) from each source are specified as follows:

CMT injection line:	$1.80 \times 10^{-5} \leq R \leq 2.26 \times 10^{-5}$ ft/gpm^2
Accumulator injection line:	$1.47 \times 10^{-5} \leq R \leq 1.83 \times 10^{-5}$ ft/gpm^2
IRWST injection line A:	$5.52 \times 10^{-6} \leq R \leq 9.21 \times 10^{-6}$ ft/gpm^2
IRWST injection line B:	$6.20 \times 10^{-6} \leq R \leq 1.04 \times 10^{-5}$ ft/gpm^2
containment recirculation line A:	$R \leq 1.12 \times 10^{-5}$ ft/gpm^2
containment recirculation line B:	$R \leq 1.04 \times 10^{-5}$ ft/gpm^2
CMT cold leg balance line:	$R \leq 7.22 \times 10^{-6}$ ft/gpm^2

Describe how these acceptance criteria are derived, and how they are consistent with the assumptions used in the safety analyses.

440.144

For Item 8.b in Tier 1 ITAAC Table 2.2.3-4, regarding the PRHR heat exchanger (HX) capability, the "Inspections, Tests, Analyses" column describes the conditions under which the PRHR HX heat removal test will be conducted, i.e., the hot leg temperature initially at $\geq$ 540°F with the RCPs stopped, and continues until the hot leg temperature decreased below 420°F. The acceptance criteria for the PRHR HX heat transfer rate is identified as "TBD Btu/hr with 520°F HL Temp and 120°F IRWST temperatures."

A. Since the IRWST water temperature can increase to the boiling temperature during the PRHR HX operation, why is it that the acceptance criteria for the PRHR HX heat transfer rate is specified for only the initial phase of the test, but not the end of the test condition? Why is one criterion sufficient to demonstrate the PRHR HX heat transfer capacity?

B. When will the PRHR HX heat removal acceptance criteria be determined?

440.145

In Tier 1 ITAAC Table 2.2.4-4, the acceptance criteria for the main steam safety valve (MSSV) (Item 8.a) is shown as 8,300,000 lb/hr per SG, and the acceptance criteria for the main steam power-operated relief valve (PORV) (Item 9.b) is 300,000 lb/hr at 1106 psia ± 10 psi.

Explain how these acceptance criteria are consistent with the design data shown in Tier 2 Table 10.3.2-1, where the PORV design capacity is shown to be 70,000 lb/hr at 100 psia inlet pressure, and 1,020,000 lb/hr at 1200 psia inlet pressure, and Tier 2 Table 10.3.2-2, where the MSSV relieving capacity is shown to be 8,340,000 lb/hr per steam line at 110 percent design pressure.

440.146

Tier 1 Table 2.3.6-1 provides a list of RNS components and quality requirements, including, among others, the RNS heat exchanger (HX) - channel head drain valve RNS-PL-V046.

A. Explain why Table 2.3.6-1 does not include the HX-B channel head drain valve RNS-PL-V048.

B. Explain why the cask load pit isolation valve RNS-PL-V055, and RNS pump mini-flow air-operated isolation valves RNS-PL-V057A and -V057B do not have valve position indication.

440.147

In Tier 1 ITAAC Table 2.3.6-4, Item 9.d specifies a design commitment of the RNS to provide heat removal from the IRWST with the acceptance criteria that each RNS pump provides at least 925 gpm to the IRWST.

Describe how this acceptance criteria is established.

440.148

Tier 1 Table 2.5.1-1, “Functions Automatically Actuated by the [diverse actuation systems] DAS,” lists the functions automatically actuated by the DAS. These functions are consistent with the DAS automatic actuation functions described in Tier 2 Section 7.7.1.11, except the following two items.

- Table 2.5.1-1 indicates that the IRWST gutter isolation valve closure is actuated on low wide-range SG water level or on high hot leg temperature, whereas Tier 2 Section 7.7.1.11 indicates the IRWST gutter isolation is actuated on the high hot leg temperature signal only.

- Table 2.5.1-1 does not include the DAS function that “trips rods on motor generator set” described in Tier 2 Information.

Explain the differences between the Tier 1 and Tier 2 information regarding the above two items.

SERIES 720 - RELIABILITY AND RISK ASSESSMENT

The following questions refer to Chapter 6 and Appendix A of the Westinghouse AP1000 PRA report that was submitted on March 28, 2002.

720.002

Regarding the acceptance criteria for decay heat removal and RCS inventory, Section A2.2 states that “adequacy of core cooling is established by requiring that either the core remains covered with water or the peak cladding temperature (PCT) of the fuel is less than 2200°F at all times during an event. In addition, small core uncoveries that have an extended and slow recovery are not considered success even if the PCT is below 2200°F.”

Define and quantify “small core uncoveries that have an extended and slow recovery” for which the core cooling is not considered success even if the PCT is below 2200°F.

720.003

Figures A3.2-7, A3.2-13, and A3.2-19, which show the best estimate MAAP4 analysis results of the break sizes of 0.5, 2.0, and 5.0 inches, respectively, show core uncovery for certain periods of time. Figure A5.2-35, the NOTRUMP design-basis-analysis-like result for the thermal-hydraulic (T/H) Uncertainty Analysis Case No. 3 (UC3), shows core uncovery of 1000 seconds duration (from 1900 to 2900 seconds in the transient). Should these results be considered as extended and slow recovery to result in a failure of core cooling even if the PCT is less than 2200°F?

720.004

Section A.2.4 of Appendix A to the AP1000 PRA states that the MAAP4 code is used to justify post-automatic depressurization system (ADS) success criteria, both for short-term and long-term core cooling. The MAAP4 code has not been submitted for NRC staff review, therefore the NRC staff must review those portions of the code relevant to each application including critical models, assumptions, code input, and experimental verification.

In order that the staff may exercise the MAAP4 code for the AP1000 and evaluate the effects of the various user options selected by Westinghouse, please provide the parameter input file and the sequence input file used to describe the 3.5 inch hot leg break with manual ADS-4 actuation that was analyzed by Westinghouse using MAAP4 as described in Section A2.4.2 of Appendix A to the AP1000 PRA. Please provide this input in electronic form.

720.005

Many of the analyses described in Section A.3 performed to justify the core cooling success paths are performed using the MAAP4 code. Use of MAAP4 is based on comparisons with NOTRUMP as described in ADS-4-14869, “MAAP4/NOTRUMP Bench marking to Support Use of MAAP4 for AP600 PRA Success Criteria Analyses,” for the AP600 review. Section A.2.4.2 provides a description of MAAP4/NOTRUMP benchmark to demonstrate the acceptability of MAAP4 for use in the AP600 PRA success criteria analyses. It states that the Bench marking work provides clear definition of MAAP4 capabilities and limitations.

For the AP1000, the NRC staff has informed Westinghouse (letter from James E. Lyons to W. E. Cummins, “Applicability of AP600 Standard Plant Design Analysis Codes, Test Program and Exemptions to the AP1000 Standard Plant Design,” dated March 25, 2002) of possible deficiencies in the NOTRUMP entrainment models at the time of ADS-4 actuation. Therefore the NOTRUMP-MAAP4 benchmarks may not be valid for the AP1000. The NRC staff must therefore assess the validity of the entrainment models in MAAP4.

A. Provide justification that the MAAP4 models are appropriate for the AP1000 analyses, including comparisons to appropriate experimental data for the liquid entrainment models in MAAP4 for the reactor core, upper plenum, hot legs, and ADS-4. Justify that the predictions by MAAP4 for the AP1000 are within the range of the test data.

B. Describe the nucleate boiling heat transfer correlation used to model heat transfer between the PRHR heat exchanger (HX) tube bundle and the in-containment water storage tank (IRWST) in MAAP4. Justify that this correlation is appropriate for AP1000 by comparison to data in WCAP-12980, “AP600 Passive Residual Heat Removal Heat Exchanger Test Final Report,” Revision 3.

C. List any limitations for the application of MAAP4 to the AP1000, such as the limitation described in of WCAP-14869 for the MAAP4 application to the AP600.

720.006

The MAAP4 analyses to justify the success paths described in Sections A.3.2.1 and A.3.3.1 are for hot leg break locations. This is because analyses in WCAP-14869 indicate that hot leg breaks are more limiting than cold leg breaks of the same break size. The hot legs are at a lower elevation relative to the reactor core than the cold legs and would be expected to cause more water to be lost through a postulated break. Analyses in Chapter 15 of the DCD indicate that cold leg break sizes are more limiting. This is because the cold leg breaks analyzed in the DCD are located close to the pressure balance line of CMT causing delayed CMT injection. Please provide analyses demonstrating that the success paths of the PRA are still valid for the break locations analyzed in Chapter 15 of the DCD.

720.007

Figure A3.2-1 is titled “Minimum Core Mixture Level for Spectrum of Break Sizes (with RNS [normal residual heat removal system] Injection).” Figure A3.2-26 is titled “Minimum Core Mixture Level for Spectrum of Break Sizes (with IRWST Injection).”

Are “RNS Injection” and “IRWST Injection” in these respective titles reversed?

720.008

Figures A3.2-2 through A3.2-25 provide comparisons between the AP1000 and the AP600 results of plant responses for various break sizes. It is stated that the AP600 plant response to these cases was based on both MAAP4 and NOTRUMP analyses. Are the AP1000 results shown in these figures also based on both the MAAP4 and NOTRUMP analyses, or the MAAP4 analyses only?

720.009

Section A3.3.1 discusses success paths involving manual ADS-4 leading to IRWST gravity injection. Analyses by both MAAP4 and NOTRUMP are referenced. Successful operation of the following equipment is credited in these success paths for hot leg break sizes of 3.5 to 8.75 inches: the PRHR HX, 1 of 2 accumulators, 3 of 4 ADS-4 valves and, 1 of 2 IRWST injection paths. Containment isolation failure is assumed so that the containment remains at a low pressure throughout the event.

A. These success paths do not appear to be bounded by any of the analyses in Section A5.1 “T/H Uncertainty Cases for AP1000.” Cases 1 and 2 assume manual actuation of ADS-4; however Cases 1 and 2 assume credit for 4 of 4 ADS-4 valves and take credit for elevated containment pressure. Furthermore, Case No. 2 assumes credit for 2 of 2 accumulators. Please provide T/H uncertainty evaluation for the success paths in Section A3.3.1.

B. Credit for the PRHR HX is stated to be required for some of these breaks, mainly between approximately 3 inches and 4 inches. In evaluating experimental data from the PRHR test facility, the nucleate boiling heat transfer correlation used in NOTRUMP was evaluated and found to produce heat fluxes that were too high in comparison to the data as the PRHR heat load increased. See Section 1.11 of WCAP-14807, “NOTRUMP Final Validation Report for AP600.” Please provide an analysis for the equipment operability assumptions listed in Section A.3.3.1 and Cases No. 1 and No. 2 in Section A5.1 showing that success will still be obtained if a PRHR HX correlation is used that matches experimental data.

720.010

Section A3.4 (also Section A5.1, Large-Break LOCA Case No. 1) discusses the AP1000 success criteria of one accumulator operating for the events of spurious opening of all four ADS-4 valves. It indicates that the PCT for this equivalent of a large-break LOCA event caused by the spurious opening of all ADS-4 valves with only one accumulator available is less severe than a double-ended cold break LOCA with both accumulators available. It provides a hand calculation estimate of the PCT of 1739°F and an uncertainty of 251°F for a total of 1990°F.

A. Is the uncertainty value of 251°F a WCOBRA/TRAC analysis uncertainty value or a hand calculation estimate uncertainty value? If 251°F uncertainty is an uncertainty for your PCT estimate, what is the basis for this value?

B. If the 251°F is based on WCOBRA/TRAC analysis uncertainty, what is the basis for applying this value to your PCT estimate? To be consistent, provide the result of WCOBRA/TRAC calculation, or the analyses using approved methodology.

720.011

In the ATWS analysis described in Section A4, Figures A4.2.1-2 and A4.2.2-2 (a typographical error in the figure) for the equilibrium cycles and the first cycle, respectively, show that the liquid volume in the pressurizer reaches 2200 cubic feet (ft^3) after about 105 seconds. However, the AP1000 pressurizer design has an internal volume of 2100 ft^3.

A. Explain the inconsistency in the pressurizer volume.

B. If the pressurizer volume of 2100 ft^3 was used in the ATWS analyses, would the peak RCS pressure exceed 3200 psia for the cases analyzed?

C. On Figures A4.2.1-4 and A4.2.2-4, explain why the nuclear power begins at 20 percent rated power.

720.012

For the AP1000 T/H uncertainty evaluation, only three cases listed in Table A5.2-1 are chosen as low T/H margin, risk-significant cases. These three cases are chosen based on an evaluation, described in Section A5.1, of the applicability to the AP1000 of the the AP600 T/H uncertainty analysis cases listed in Table A5.1-1. Many of these 15 AP600 cases are determined to be not applicable to the AP1000 because of different success criteria used in the AP1000. Limiting the selection of low T/H margin, risk-significant cases for the AP1000 T/H uncertainty analysis from the AP600 low T/H margin cases could preclude other possible low T/H margin, risk-significant cases for the AP1000.

A. Provide a systematic evaluation of the AP1000 PRA accident sequences to choose the low thermal margin, risk-significant cases for T/H uncertainty evaluation.

B. Provide an evaluation of these cases using the design basis accident (DBA)-like analyses to demonstrate the acceptability of the PRA success criteria.

720.013

For post-ADS long-term cooling, Section A3.5 (and Section A5.1 Long-Term Cooling Cases No. 5 and 6) provides your judgement, based on the AP1000's increased power level, ADS-4 flow capacity, IRWST injection, and containment recirculation over the AP600 design. The conclusion provided is that the AP1000 ADS-4 vent capacity is sufficient for justification of the long-term core cooling success criteria assumed in the AP1000 PRA Section A5.2 states that Section A5.5.2 documents the long-term cooling analyses performed with the WCOBRA/TRAC code with the details of the analysis methodologies used provided within each subsection. However, there is no Section A5.5.2.

Provide the WCOBRA/TRAC long-term cooling analysis as part of the AP1000 PRA to support your conclusion.

720.014

Section A5.1 indicates that Westinghouse's purpose is to bound the T/H uncertainly for the various success paths rather than to quantify it. For Case UC3 (a double-ended DVI line break for which the accumulators are assumed to have failed) two of the assumptions may not be bounding. Please supply supporting analyses to demonstrate that bounding assumptions have been made.

A. The CMT isolation valve on the broken DVI side is assumed to have failed closed. If the CMT valve was assumed to open as designed, earlier ADS-4 actuation would occur from draining of the affected CMT but more coolant would be lost. Justify that to fail the isolation valve closed is bounding.

B. The IRWST injection valve on the broken DVI side is assumed to have failed closed. If the IRWST valve was assumed to open as designed, IRWST water would drain on the containment floor lowering the IRWST water level and delaying the time when water would begin to enter the reactor through the intact DVI line. Justify that to fail the IRWST injection valve closed is bounding.

720.015

Section A5.2.1.1 states for cases where containment isolation is assumed, a constant containment back pressure of 25 psia is used. This is for the 3.25 inch hot leg break and the double-ended CMT inlet breaks with manual ADS-4 actuation.

Provide a containment pressure analyses justifying that this assumption is bounding for the entire duration of the events, in particular for the period between ADS-4 actuation and IRWST injection.

720.016

Section A5.2.1.3.3 describes the results from T/H Uncertainly Case No. 3 (UC3) as an estimated cladding heat-up of well less than 2000°F. Section A5.2.1.2 states that the cladding heat-up calculations were made using the LOCTA code.

A. Describe how this calculation was made and why the results are considered to be only estimated values as stated on page A-36.

B. Provide the PCT calculated by LOCTA as a function of time.

720.017

Uncertainty Case No. 3 (UC3) discussed in Section A5.2.1.3.3 cannot be considered a success path because long-term cooling capability has not been established. Success in long-term cooling is referenced to the analysis in Chapter 15 of the DCD (4-out-of-4 ADS-4 valves operable). The Chapter 15 analysis takes credit for the increase in containment pressure that would occur as the result of containment isolation. Containment isolation is not assumed for case UC3 and only three of four ADS-4 valves are assumed to be operable.

Provide a long-term cooling T/H uncertainty evaluation for the set of equipment assumed to be operable for case UC3.

720.018

Comparisons with WCOBRA/TRAC in WCAP-15833, “WCOBRA/TRAC AP1000 ADS-4/IRWST Phase Modeling,” indicate that NOTRUMP is non-conservative for calculating reactor vessel inventory. See Figures 3-16 and 3-25 of the WCAP. The reduced vessel inventories are a result from more detailed models for reactor vessel, hot leg and ADS-4 entrainment in WCOBRA/TRAC.

Provide sensitivity studies showing the effect on PCT if the more detailed models of WCOBRA/TRAC are used to calculate reactor vessel inventory for Uncertainty Case Nos. 1, 2, and 3 in Section A5 of the PRA.

720.019

The T/H uncertainty evaluation in Section A5 utilized the NOTRUMP code to evaluate minimum sets of equipment needed to prevent core damage. The NOTRUMP code was qualified for the AP600 by comparisons of code predictions to test data from integral and separate effects test facilities as discussed in WCAP-14807. Following the pre-application review for the AP1000, the NRC staff informed Westinghouse that "none of the integral effects facilities are sufficiently well scaled so that they provide an acceptable data base to validate T/H codes for the high rate of liquid entrainment that are expected to occur in the AP1000 during ADS-4 and IRWST injection periods of a small-break LOCA—The staff concludes that Westinghouse must either obtain entrainment test data applicable to the AP1000 steam flow rates for code verification or provide justification for the entrainment models to be used for the AP1000 applications."

Provide an evaluation showing the effects on core cooling for the various small-break LOCA success paths resulting from the uncertainty in ADS-4 liquid entrainment.

720.020

As described in Section A5.2.1.2, the LOCTA code is used for cladding heatup calculations in the PRA when noticeable core uncovery is predicted. The LOCTA code has been approved by the NRC staff for core heatup analyses for operating reactors. The high void fractions and low pressures predicted within the core of the AP1000 following ADS-4 activation may lie outside the range of the critical heat flux and transition boiling correlations contained in the code. In its March 15, 2002, letter to Westinghouse, the staff identified the need for the justification of the methodology used to calculate PCT in the event that the core becomes uncovered during a small-break LOCA.

A. Provide the pressure and void fraction limits for these correlations and demonstrate that the correlations are being used within these limits.

B. Since the AP1000 PRA small-break LOCA result in core uncovery, provide further justification why the LOCTA code is applicable for the AP1000 PRA analyses.

720.021

Section 6.3.1.5, "Containment Isolation," states that "analyses (documented in Appendix A) were conducted to show that sufficient water for long-term cooling of the core will be retained in containment even if containment isolation is unsuccessful."

A. Clarify where in Appendix A these analyses are documented.

B. Provide the supporting analysis, including all assumptions with justifications, to justify the conclusion that sufficient water for long-term recirculation core cooling will be retained in the containment without containment isolation. The analysis should be specific to the AP1000 rather than reference to the AP600 analysis since the AP1000 operates at a higher power with more decay heat and could be expected to lose more coolant through an unisolated containment by steaming than would the AP600.

720.022

Section 6.3.4 states “as discussed in Appendix A, the effect of failure to isolate the containment is considered in all success trees.”

Clarify this statement since two of the success paths evaluated in Section A5 of Appendix A take credit for containment isolation and the resulting increase in reactor system pressure.

720.023

Section 6.4 states “The entries in Table 6-1 for ‘Basis’ refer to specific subsections of Section 6.3.” The cross references to these subsections appear to be omitted from Table 6-1. Please provide them.

720.024

Table 6-1 provides a “Summary of Event Tree Top Events Success Criteria.” For each event case a basis for success is provided and referred to as “Calculated value,” “Design Basis,” “Provided for sensitivity studies,” “PRA specific analysis,” “Other specific justification,” “Engineering judgment,” or “Other transient analysis.”

For each event case in Table 6-1, please provide or give specific reference to the analysis or justification that is the basis for declaring the event case to be a success for the AP1000. If the justification involves analyses for the AP600, justify that the analysis is applicable to the AP1000.

720.025

Section 6.3.2 discusses key operator actions within the various accident sequences, and the available time for these operator actions. For example, Section 6.3.2.1 states that for a loss of feedwater event, the maximum time available for manual PRHR actuation is determined to be greater than 45 minutes. Section 6.3.2.5 states that, for a medium-break LOCA, the time available for operator action to actuate CMT injection is determined to be 10 minutes from the time the actuation signal occurs, and 20 minutes if accumulator injection is successful. With successful accumulator injection and PRHR operation, the available time for operator action to depressurize the RCS is determined to be approximately 20 minutes from the time CMT actuation occurs, and the time available to start RNS injection is determined to be 20 minutes. For a small-break LOCA, SG tube rupture and transients, the time available to manually actuate CMT, and RCS depressurization is determined to be at least 30 minutes. The maximum time available for manual actuation of RNS is determined to be approximately 10 minutes from ADS actuation if PRHR has not actuated. In all cases, Appendix A is referenced for the determination of the maximum available time for the operator actions.

A. Clarify where in Appendix A these available operator action times are described.

B. Provide bases and determination of these available times for the AP1000 design.

720.026

Section 6.3.3.1 discusses the impact of the Chemical and Volume Control System (CVS) operation on depressurization, gravity injection, and PRHR operation. It states that PRA success criteria analysis sensitivity to numbers of CMTs and accumulators bounds the impact of possible CVS operation and references Appendix A for the sensitivity analysis.

A. Clarify where in Appendix A these sensitivity analyses are described.

B. Are the sensitivity analyses performed for the AP1000 design?

<u>HISTORY OF PREVIOUSLY-ISSUED</u>
<u>REQUESTS FOR ADDITIONAL INFORMATION</u>

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		
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		

Enclosure 2

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