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

May 2, 1994

Docket No. 52-003

Mr. Nicholas J. Liparulo
Nuclear Safety and Regulatory Activities
Westinghouse Electric Corporation
P.O. Box 355
Pittsburgh, Pennsylvania 15230

Dear Mr. Liparulo:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION ON THE AP600

As a result of its review of the June 1992 application for design certification of the AP600, the staff has determined that it needs additional information in order to complete its review. The additional information is needed in the area of reactor systems (Q440.85-Q440.121).* In addition, we request that Westinghouse provide a complete set of up-to-date piping and instrumentation diagrams (P&IDs) (Q100.12). Enclosed are the staff's questions. Please respond to this request by June 30, 1994 to support the staff's review of the AP600 design.

You have requested that portions of the information submitted in the June 1992 application for design certification be exempt from mandatory public disclosure. While the staff has not completed its review of your request in accordance with the requirements of 10 CFR 2.790, that portion of the submitted information is being withheld from public disclosure pending the staff's final determination. The staff concludes that this request for additional information does not contain those portions of the information for which exemption is sought. However, the staff will withhold this letter from public disclosure for 30 calendar days from the date of this letter to allow Westinghouse the opportunity to verify the staff's conclusions. If, after that time, you do not request that all or portions of the information in the enclosures be withheld from public disclosure in accordance with 10 CFR 2.790, this letter will be placed in the NRC's Public Document Room.

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

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

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

Sincerely,

(Original signed by)

Thomas J. Kenyon, Project Manager
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Associate Director for Advanced Reactors
and License Renewal
Office of Nuclear Reactor Regulation

Enclosure:
As stated

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REQUEST FOR ADDITIONAL INFORMATION
ON THE WESTINGHOUSE AP600 DESIGN

GENERAL

100.12 Several piping and instrumentation diagrams (P&ID) are not self-contained. For example, the passive core cooling system P&IDs in Figures 6.3-1 through 6.3-4 of the SSAR contain many interfaces with other systems whose P&IDs are not provided. Provide a complete set of up-to-date P&IDs for the AP600 design.

REACTOR SYSTEMS

440.85 In the discussion of the passive core cooling system (PXS) design basis for emergency core makeup and boration for non-LOCA events, Section 6.3.1.1.2 of the SSAR states that following either small or large steam line break events, the RCS is automatically brought to a subcritical condition, consistent with the passive containment cooling capabilities. Clarify the relationship between subcriticality and the passive containment cooling capabilities, and the phrase "automatically."

440.86 Section 6.3 of the SSAR states that the emergency core heat removal function of the PXS system is available at reactor coolant system conditions including shutdown and refueling. Is this statement true for refueling conditions when most of the IRWST water is transferred to the refueling pool and not available for the PXS system operation?

440.87 Section 6.3.1.1.2 of the SSAR states that for safe shutdown, the passive core cooling system is designed to supply sufficient boron to the RCS to maintain the Technical Specification (TS) requirements for shutdown margin (with the CVCS unavailable). TS LCO 3.1.1 and 3.1.2 specify that the required shutdown margins for various modes of plant conditions, and if they are not met, require the operator to initiate boration to restore shutdown margins to within the specified limits without identifying the boron injection source, i.e., the CVCS or the CMT.

a. Discuss the required boron concentrations to maintain the shutdown margin for cold and post-depressurization conditions, and the source of boration to be used to implement LCO 3.1.1 and 3.1.2.

b. Describe how boron injection to the RCS is accomplished using the passive core cooling system alone.

c. Identify any deviations from the boron concentrations in current operating plant requirements.

Enclosure

440.88 Section 6.3.1.1.6 of the SSAR discusses the PXS reliability requirements, and states that "Subsection 6.3.1.2 includes specific non-safety-related design requirements" that help to confirm satisfactory system reliability. Section 6.3.1.2 does not appear to address these design requirements. Provide a discussion of these specific non-safety-related design requirements.

440.89 Section 6.3.1.2 of the SSAR states that the frequency of automatic depressurization system actuation is limited to a low probability to reduce safety risks and to minimize plant outage. Define the term "low probability," and explain how this goal is achieved.

440.90 The PXS is designed to supply the core cooling flow rates to the RCS during accidents. Section 6.3.2.1 of the SSAR states that the Chapter 15 accident analyses flow rates and heat removal rates are calculated by assuming a range of component parameters, including best estimate and conservatively high and low values. Clarify when the best estimate values are used in the licensing design basis analyses, and provide the bases for their use.

440.91 Section 6.3.2.1.1 of the SSAR states that the PRHR heat exchangers are connected to the RCS through a common inlet line from one RCS hot leg (through a tee from one of the fourth stage ADS lines) and a common outlet line to the associated steam generator cold leg plenum. Because a common mode failure (e.g., a break of the common lines) would disable both PRHR heat exchangers, discuss the reasons for this arrangement.

440.92 Sections 6.3.1.1.4 and 6.3.2.1.1 of the SSAR state that the passive residual heat removal (PRHR) heat exchangers, in conjunction with the passive containment cooling system (PCCS), has the capability to bring the plant to safe shutdown conditions, cooling the RCS to about 400°F in 72 hours for non-LOCA events, and can provide core cooling for an indefinite period of time based on the assumption that the PCCS is operable for indefinite period of time to promote condensation of the steam from and return the condensate to the IRWST.

a. Is this PRHR system capability of cooling RCS to 400°F in 72 hours consistent with the requirement specified in the EPRI ALWR Utility Requirements Document for passive plants that the PRHR system shall have sufficient capacity to reduce coolant temperature to 420°F within 36 hours of reactor shutdown?

b. The PCCS is designed with a 72-hour capacity, and relies on non-safety equipment to replenish the lost water in the reservoir beyond 72 hours. Justify Westinghouse's position that the PCCS will be operable for an indefinite period of time.

c. For the condensate to return directly to the IRWST, an isolation valve in each of the two gutters that normally drain to the containment sump will be shut when the PRHR heat exchangers actuate. What is the design requirement of the isolation valve to ensure its closure upon PRHR actuation?

d. The IRWST water inventory is sufficient to provide the PRHR heat exchanger operation for 72 hours without recovery of the condensate. Will the heat exchangers remain totally submerged in the IRWST water to maintain full operational capability for 72 hours? If not, how are the progression of the uncovery of the heat exchangers during transients and the degradation of the heat removal capacity accounted for in the safety analyses?

440.93 What considerations and analyses have been made to the PRHR system design to accommodate thermal transient, thermal stress, and fatigue associated with system initiation where the fluid temperature will change from cold containment ambient condition to the hot leg temperature within a short period of time? What are the number of operating cycles that the PRHR system is designed for?

440.94 What considerations and analyses have been made in the design of the steam generator channel head injection nozzle to accommodate thermal stress due to PRHR operation as well as other thermal transients, and to ensure thermal fatigue due to high frequency thermal cycling is acceptable over the 60 year plant life?

440.95 Due to concerns of adverse systems interactions (ASI) among various structures, systems, and components in a plant, USI A-17 identifies the need to investigate the possibility that unrecognized subtle dependencies among SSCs have remained hidden and could lead to safety significant events. The staff efforts on USI A-17 resulted in the issuance of NUREG-1174, NUREG-1229, Generic Letter 89-19, and SECY-89-230. Technical findings described in NUREG-1174 include (a) categorization of intersystem dependencies based on the way they propagate into functionally-coupled, spatially-coupled, and induced human-intervention coupled systems interactions, and (b) available methods for identifying systems interactions, such as operating experience reviews, plant walk-throughs, pre-operational testing, failure modes and effects analyses, and PRA. For the operating plants, the staff did not recommend that licensees conduct further broad searches specifically to identify all ASIs because of considerations of cost-effectiveness and other ongoing activities, such as individual plant examinations, that could reduce the risk from ASIs. However, the staff also concluded that the 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, the staff believes that the designer should perform a systematic search for ASIs, and propose resolutions of any that are discovered.

As part of the evaluation of the issue of the regulatory treatment of nonsafety systems (RTNSS), an evaluation of the ASIs in the AP600 design is included in Revision 0 of WCAP-13856, with the conclusion that no non-safety-related SSC functions are relied upon to prevent the potential for the non-safety-related SSCs to adversely interact with the safety-related systems. The discussion of this matter does not provide sufficient information for the staff to evaluate this acceptability of this conclusion. For example, it is not clear how the functionally-coupled ASIs can be discovered or accounted for simply by (a) considering a non-safety-related system operation in the design basis analysis if its function worsens the analytical results, and (b) modeling the effects of system interdependencies in the system fault trees and event trees. The safety analyses and assessment may only account for those systems interactions that are readily recognized.

a. Describe in detail what considerations, evaluations, tests, and methods, including PRA, were used to systematically search for and identify ASIs in the AP600 design. This discussion should address how the operation and functional capability of the passive safety-related systems could be adversely affected by the failure or operation of the active non-safety-related systems, operator actions or errors, and other plant hazards and natural phenomena (including water intrusion, internal flood, and other coupling mechanisms such as seismic events and piping ruptures).

b. Describe the ASIs identified from the evaluation requested in Item a above, and propose resolution actions to reduce these ASIs, including design modifications and operational and emergency procedure guidelines.

c. Identify items to be included in the AP600 ITAAC program for conducting walkdowns of "as built" facilities as a resolution of the functionally- and spatially-coupled ASIs.

440.96 Section 6.3.2.2.3 of the SSAR states that an overflow is provided from the IRWST to the refueling cavity to accommodate volume and mass increases during PRHR heat exchanger and ADS operation to minimize the floodup of the containment. Clarify under what plant conditions, other than refueling, is the refueling cavity allowed to flood. How does the flooding of the refueling cavity affect thermal stress on the reactor vessel head region and the refueling cavity seal?

440.97 Section 6.3.2.2.3 of the SSAR states that flow out of the IRWST during the injection mode includes conservative allowances for spill flow during a direct vessel injection line break.

a. Describe these conservative allowances.

b. Has the bypass phenomena of the safety injection through the DVI been properly modeled in the analysis codes, and verified and validated by appropriate testings?

440.98 During transients or accident events, the core makeup tank injects cold water through the direct vessel injection line. What analyses or tests have been performed to demonstrate that the thermal and vibration effects of direct vessel injection on the reactor vessel or reactor internals are acceptable?

440.99 Sheet 1 of Table 6.3-1 of the SSAR for Case 1, "Non-LOCA, CMT Operation in Water Circulation Mode," indicates that the fluid for the direct vessel injection line (Location 35) is air. This appears to be incorrect. Should the fluid be water?

440.100 Table 6.3-4 of the SSAR provides the normal-, actuation-, and failed-positions, respectively, of the remotely actuated valves used by the various passive core cooling system components.

a. There are many valves which are normally closed and whose opening positions are important in the prevention/mitigation function of the PXS, such as the ADS MOVs and the sump recirculation line MOVs. Explain why these valves are designed to "fail as is" rather than move to the "fail safe" positions, e.g., move to the open position, even though the failure modes and effects analysis in Table 6.3-6 indicates that there is no safety-related effects for "fail as is."

b. Section 6.3.2.2.7.4 of the SSAR states that these motor-operated isolation valves have various interlocks, automatic features, and position indications. Provide a description of the interlocks and the bases for these interlocks.

440.101 General Design Criterion (GDC) 4 requires that structures, systems, and components important to safety be designed with appropriate protection against dynamic effects, including the effect of discharging fluid and loads, e.g., water hammer. WCAP-13054 states that the AP600 passive core cooling system is designed to prevent damaging water hammer by incorporating specific design features that preclude water hammer, such as sloping lines or maintaining pressure in standby components. Therefore, Westinghouse has indicated that it meets GDC 4 as related to dynamic effects associated with flow instabilities and loads. Several, but not all, of the design features are discussed in Section 6.3.2 of the SSAR, e.g., the CMT inlet diffuser design reduces steam velocities entering the CMT, thereby minimizing potential water hammer (Section 6.3.2.2.1); the depressurization spargers in the IRWST prevent undesirable and/or excessive dynamic loads on the IRWST and other structures (Section 6.3.2.2.6); and proper initial filling and venting of the PXS prevents water hammer from occurring in the PXS lines (Section 6.3.2.5). For completeness, provide a description of each of the PXS equipment and component design features that demonstrate compliance with GDC 4.

440.102 Sheet 3 of Table 6.3-6 of the SSAR, "Failure Modes and Effects Analysis for PXS Active Components," indicates that spurious opening of the accumulator nitrogen supply/vent valves has no safety-related effect since each valve has either a normally-closed, redundant series isolation SOV or a check valve in each vent flow path that prevents accumulator nitrogen from leaking out of the accumulator, which could preclude accumulator injection. Clarify or provide a more detailed P&ID that shows such a valve arrangement.

440.103 Failure of each of the isolation AOVs in the IRWST condensate return lines to close will drain the condensate to the containment sump instead of the IRWST. This will reduce the amount of condensate that returns to the IRWST. Sheet 5 of Table 6.3-6 of the SSAR states that the safety analysis has demonstrated that successful core cooling is achieved without IRWST condensate return. As stated in Section 6.3.2.1.1 of the SSAR, without the recovery of the condensate, the IRWST inventory will be sufficient to support operation of the PRHR heat exchanger for 72 hours. Was the safety analysis performed assuming operation of the PRHR heat exchangers for only 72 hours? What would be the effect after 72 hours?

440.104 Section 6.3.2.5.1 of the SSAR states that the PXS has been "specifically designed to treat check valves failures to reposition as active failures." The core makeup tank discharge line contains two tilt-disc check valves in series. The FMEA in Table 6.3-6 does not consider the failure modes for these check valves because they are not considered active failures as they are normally open and remain in the same position on demand. However, for an accident where the accumulators discharge into the RCS, these check valves will close to prevent backflow into the CMT, and will have to reopen to inject borated water into the RCS.

a. The arrangement of two check valves in series does not meet the single failure consideration. Either modify the CMT discharge line check valve arrangement, or provide justification for treating these check valves as passive components. Also, provide the results of the FMEA analysis for these CMT discharge line check valves.

b. Describe the CMT discharge line check valve design to discuss how they are normally maintained open.

c. Technical Specification 3.5.2 specifies the LCO and Surveillance Requirements (SR) for the CMT. Why are there no SRs to verify that the CMT outlet check valves are open, and no action requirement when the check valves are not open?

440.105 Section 6.3.2.5.2 of the SSAR states that the PXS can sustain a single passive failure during the long-term phase and still retain a flow path to the core to supply sufficient flow to keep the core covered and to effect the removal of decay heat. What analyses have been made to confirm this conclusion? Describe.

440.106 Section 6.3.2.5.3 of the SSAR states that for those valves that reposition to initiate safety-related system functions, the valve repositioning times are less than the times assumed in the accident analyses. It further states that it is acceptable for the CMT injection to be delayed several minutes due to high initial steam condensation rate.

a. The proposed Technical Specifications do not provide a definitive requirement regarding the valve repositioning time. For example, Surveillance Requirement 3.5.2.4 specifies verification of the CMT inlet and outlet isolation valves to be operable every 92 days without defining the valve repositioning times or what constitute operability of the valves. Describe how the valve delay times are controlled in the TSs and how surveillance is made to ensure the actual delay times are shorter than assumed in the safety analysis?

b. Describe how the CMT injection delay time is accounted for in the safety analysis and what verification is performed to ensure that this is a conservative value.

440.107 During an inadvertent opening of a steam generator relief or safety valve and steam system pipe failure, the reactor is tripped and the core makeup tank is actuated to inject borated water by the safeguards actuation signal. Sections 6.3.3.1.1 and 6.3.3.1.2 of the SSAR both state that, although the borated water solution does not provide sufficient negative reactivity to maintain the reactor subcritical, the core is ultimately shut down by the borated water solution. Explain how the reactor is shutdown by a boron solution that does not provide sufficient negative reactivity to maintain the reactor subcritical.

440.108 An AP600 design change (DM-01) (design change report dated February 15, 1994) was made to revise the PRHR heat exchanger actuation logic by adding a CMT actuation signal and deleting the high pressurizer level and high steam generator level signals. For a steam generator tube rupture event, the PRHR system is actuated by the actuation of the CMT, which is actuated on low pressurizer level, as discussed in Section 6.3.3.3.1 of the SSAR. However, Section 15.6.3.1.3 states that the AP600 steam generator overfill "protection system," actuated at the High-2 SG narrow range level setpoint, will automatically provide safety-related actions to initiate the PRHR system heat exchangers, isolate the CVCS pump, and isolate the startup feedwater pump.

a. What is the effect of this design change to the automatic actuation of the PRHR system by the SG overfill "protection system" that is actuated at high SG level?

b. What is meant by "safety-related actions" to initiate the isolation of the CVCS pumps and the startup feedwater pumps? Are these isolation valves safety-related components?

c. Is the design basis for the overfill protection system for single-tube or multiple-tube rupture events?

440.109 Section 6.3.1.1.1 of the SSAR indicates that, during a steam generator tube rupture event, the passive residual heat removal (PRHR) heat exchangers remove the core decay heat to reduce the reactor coolant system (RCS) temperature and pressure sufficiently to equalize with steam generator pressure and terminate break flow, without overfilling the steam generator. Clarify whether this PRHR system capability design basis is for single- or multiple-tube rupture events.

440.110 One of the staff concerns related to an SGTR event is the likelihood of a steam generator safety valve (SGSV) lifting and then failing to close. Such an occurrence results in an unisolable release to the environment bypassing the containment. In SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated April 12, 1993, the staff stated its position that an applicant for a passive PWR design certification should assess features to mitigate the amount of containment bypass leakage that could result from SG tube ruptures. SECY-93-087 also recommended certain design features that could mitigate the releases associated with an SGTR. Perform an evaluation and, if necessary, propose the addition or modification of mitigating features to reduce or prevent SGSV challenges during an SGTR event, including multiple tube ruptures.

440.111 Section 6.3.3.3.2 of the SSAR defines a loss of coolant accident as a rupture of the RCS piping that results in a decrease in the RCS inventory that exceeds the flow capability of the normal makeup system. However, because the AP600 normal makeup system is a non-safety-related system, credit for its makeup capability should not be taken to compensate for the loss of coolant. Appendix K to 10 CFR Part 50 requires consideration of a spectrum of possible pipe breaks. Either confirm that the small break LOCA analysis is extended to break sizes within normal makeup capability, i.e., less than 0.375-inch diameter hole, or provide justification (other than makeup capability of non-safety systems) for not evaluating this small break size.

440.112 Section 6.3.3.4.2 of the SSAR states that, with a loss of the normal RHR system when the RCS pressure boundary is intact, the PRHR heat exchangers provide the safety-related heat removal path, and that the PRHR heat exchangers can remove sufficient heat to maintain the RCS within the NRHRS design limits (400°F) and permit the NRHRS to be placed back in operation when it becomes available.

a. How long can the NRHRS be out of service while relying on the PRHR system to maintain the RCS at 400°F?

b. Provide a safety analysis of LOCA and other transient events that may be initiated at this condition.

440.113 In the discussion of the loss of the NRHRS during mid-loop operation, Section 6.3.3.4.3 of the SSAR states that the IRWST isolation valves automatically open via a signal from diverse actuation system, after a delay, if the non-safety-related RCS hot leg level indication decreases below an established setpoint. This section also states that the operator can remotely open the CMT and accumulator isolation valves to provide additional makeup water injection, if required.

a. What is the justification for crediting the non-safety-related hot leg level indication to cope with a loss of the NRHRS?

b. The AP600 design basis is to use the passive safety systems and preclude operator action for 72 hours. What is the basis for crediting operator action during a loss of the NRHRS?

c. Provide a safety analysis of the loss of the NRHRS during mid-loop operation taking no credit for the non-safety related systems or components, nor operator actions

440.114 Section 6.3.3.4.4 of the SSAR states that in the event of a loss of the NRHRS during refueling, if the containment is not sealed, boiling will reduce the refueling cavity water level to the top of the fuel in about 5 days, but continued core cooling can be easily maintained by one of several methods, such as closing the containment and using multiple non-safety-related systems. Because there is no Technical Specification control for the non-safety systems (e.g., CVCS), what is the basis or justification to assume their availability?

440.115 Sections 6.3.7.7 and 7.4 of the SSAR state that there is a timer that automatically actuates the automatic depressurization system if the offsite and onsite ac power sources have been lost for about 24 hours, that the dc batteries that power the ADS valves provide power for at least 24 hours, and that the ADS valves will be opened before the batteries are discharged. Provide the detailed sequencing and timing of the actuation of various ADS stage valves after a loss of the ac power source.

440.116 TS LCO 3.5.3 specifies requirements for the PRHRS. Required Action Item D specifies that, if the other required actions for PRHRS operability are not met, immediately initiate actions to restore the PRHR to operable status.

a. How long is the allowed outage time to complete restoration of PRHRS operability? What is the basis for this allowed outage time?

b. Revision 1 of the TSs deletes the required action D.2 in the original TS for evaluation of the availability of alternative heat removal paths within one hour. Why is this action not deleted from the BASES? BASES D.2 states that if the PRHRS has been determined to be inoperable and the plant is operating with the main feedwater system supplying the steam generators, and if an alternate heat removal path, such as the startup feedwater system, and the steam generator are operable and would be available when required, then the PRHRS may not be needed during the initial phase of cooldown. Because the startup feedwater system is a non-safety-related system and not subject to TS control for operability, how can it be assured that it will be available when required? What constitutes the operability of the startup feedwater system? What is the basis for taking credit for a non-safety related system in determining the allowed outage time of a safety-related system?

440.117 TS LCOs 3.4.12, 3.5.2, 3.5.3, and 3.5.4 require the ADS, both core makeup tanks, PRHRS, and IRWST, respectively, to be operable during plant operation modes 1 through 5. If the LCOs are not met, and all the corrective actions fail, LCOs 3.4.12 and 3.5.2 require the plant to enter LCO 3.0.3 immediately, i.e., the plant should be placed in a safer mode or other specified condition in which the LCO is not applicable. LCO 3.5.4 requires the plant to be in a mode to be determined if all corrective actions fail. Clarify what mode of safe operation the plant should be in if corrective actions failed to restore the ADS, CMT, PRHRS, and IRWST within their respective LCOs.

440.118 In the TS sections regarding the passive safety systems, e.g., LCO 3.4.12 for the ADS, and LCO 3.5 for the PXS, a majority of the required action completion times and SR frequencies are not given, but indicated "TBD." Section 16.1.1 of the SSAR indicates that "TBD" is for those cases where the detailed design, equipment selection, or other efforts are not sufficiently complete to establish the information required to be specified in TS, and that some of the information, such as the established startup testing, will not be available until a plant is constructed.

a. These allowed outage times and surveillance frequencies are important information in the staff's evaluation of the acceptability of these safety systems and their associated TSs. Define which items defined as "TBD" are due to the lack of a detailed design that will not be available until the plant is constructed, and which items will be completed before plant construction, and the date that they will be provided for staff review.

b. For those items that are not available prior to construction, provide bounding values.

c. Because modifications have been made to the PXS system component (e.g., ADS) designs as described in the AP600 Design Change Description Report, dated February 15, 1994, revise the corresponding TSs (e.g., TS 3.4.12) to be consistent with the new design.

440.119 Several modifications have been made to the ADS valve design as described in AP600 Design Change Description Report, dated February 15, 1994. These changes allow flexibility to use different types of ADS valves, ranging from globe valves to squib valves. Section 2.9 of the report states that an AP600 safety analysis will be performed using bounding ADS valves and system parameters. Discuss when the final design of the various stages of ADS valves will be finalized, and how the Tier 1 ITAAC will be affected.

440.120 The staff is concerned with boron dilution events for PWR designs. A slow, inadvertent dilution due to malfunction of the chemical and volume control system (CVCS) or faulty operator actions is a design basis event that must be shown to satisfy stringent acceptance criteria. Recently, the question of whether additional failures or scenarios other than the CVCS malfunction events might lead to inadvertent criticality and fuel damage has received considerable attention in Europe and the United States. For example, a preliminary study by the Finnish Center for Radiation and Nuclear Safety indicates that an inherent mechanism for boron dilution exists in the cold leg loop seal for transients and accidents, e.g., a small break LOCA, involving heat removal by reflux- or boiler-condensation natural circulation. Under certain conditions and scenarios, such as during the restart of RC pumps, substantial boron dilution could result in the core, leading to a reactivity induced accident.

a. Although the AP600 design does not have a loop seal in the cold leg, has Westinghouse evaluated the possibility of accumulating deborated (a highly diluted slug) water in the reactor coolant loop, especially in the steam generator cold leg channel head, as a result of reflux/boiler condensation natural circulation in an accident? Address this concern.

b. For those transients or accidents that may result in the accumulation of a deborated water slug in the RCS loop, provide an analysis to demonstrate that recriticality will not occur as a result of the deborated water slug entering into the core, either through natural circulation or by restarting the pump(s). The analysis should include an evaluation of the degree of mixing between the deborated water slug and the existing borated water in the core and/or elsewhere (downcomer, lower plenum, etc.), the resulting boron concentration, the reactivity insertion, and the total reactivity. Describe the methodologies used in the analysis.

c. If recriticality occurs, provide an analysis of the consequence, such as whether the calculated peak fuel enthalpy (due to insertion of reactivity) has exceeded the limiting value of 280 calories per gram.

d. What emergency operating procedures are there to prevent the restart of RC pump that could result in criticality during transients and accident events? What are other protective measures?

440.121 Generic Issue 122.2 deals with the adequacy of emergency procedures, operator training, and available monitoring systems for determining the need to initiate feed-and-bleed cooling following loss of the steam generator heat sink. The AP600 design relies on feed (from the IRWST, CMT, and accumulator) and bleed (through the ADS) operation as backup to the startup feedwater and PRHR heat exchanger. Provide a discussion of the emergency procedure guidelines, operator training, parameters, and instrumentation and control systems relevant to the initiation of feed and bleed.