September 11, 2003

MEMORANDUM TO: Laura A. Dudes, Section Chief
New Reactors Section
New, Research and Test Reactors Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

FROM: Joseph Colaccino, Senior Project Manager ***/RA/***
New Reactors Section
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Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

SUBJECT: JANUARY 13, 16, AND 29, AND FEBRUARY 13, 2003, TELEPHONE CONFERENCE CALLS SUMMARY

On Monday, January 13, 2003, Thursday, January 16, 2003, Wednesday, January 29, 2003, and Thursday, February 13, 2003, telephone conference calls were held with Westinghouse Electric Company (Westinghouse) representatives and Nuclear Regulatory Commission (NRC) staff to discuss responses to several requests for additional information (RAIs). The following RAIs were discussed: 440.014, 440.022, 440.037, 440.043, 440.048, 440.049, 440.050, 440.052, 440.076, 440.077, 440.089, 440.091, 440.092, 440.097, 440.099, 440.102, 440.119, 440.183, and 440.184, 720.005, 720.009, 720.010, 720.012, 720.013, 720.017, 720.021, 720.024, and 720.025 (NOTE: RAI 440.183 was sent to Mr. Michael Corletti of Westinghouse via electronic mail on January 28, 2003, and issued in a letter dated February 5, 2003 [RAI 440.184 was discussed with Westinghouse in the January 28, 2003, call but no information was sent to Westinghouse about the RAI until it was issued via letter dated February 5, 2003 ADAMS Accession No. ML030340065]). Westinghouse submitted responses to these RAIs on October 2, 2002 (ADAMS Accession No. ML022810450), October 18, 2002 (ADAMS Accession No. ML022980577), November 1, 2002 (ADAMS Accession No. ML023110249), November 8, 2002 (ADAMS Accession No. ML023170535), November 15, 2002 (ADAMS Accession No. ML023230385), November 26, 2002 (ADAMS Accession No. ML023360097), and December 2, 2002 (ADAMS Accession No. ML023400058). In addition, the following Westinghouse topical reports which were submitted on December 2, 2002, were discussed: WCAP-15985, “AP1000 Implementation of the Regulatory Treatment of Non[-]safety-Related Systems [RTNSS] Process,” WCAP-15992, “Adverse System Interactions Evaluation Report,” and WCAP-15993, “Evaluation of the AP1000 Conformance to Inter-System Loss-of-Coolant Accident Acceptance Criteria.” These topical reports were submitted on December 2, 2002 (ADAMS Accession Nos. ML023370584, ML023370619, and ML023400402, respectively) in response to several NRC staff RAIs. The purpose of the telephone conference call was to discuss with Westinghouse representatives those RAI responses that require further clarification. A list of participants is included in Attachment 1. Attachment 2 contains NRC staff comments regarding the subject RAIs that were sent to Mr. Michael Corletti of Westinghouse via electronic mail on January 10,

13, 14, 28, and February 11, 2003, that were used to facilitate discussions during the telephone conference calls.
Following is a brief summary of the discussions regarding the identified RAIs (see comments in Attachment 2):

<u>WCAP-15993, "Evaluation of the AP1000 Conformance to Inter-System Loss-of-Coolant Accident Acceptance Criteria"</u> (discussed on January 13, 2003)

In response to the comments, Westinghouse representatives stated that WCAP-15993 would be revised as follows:

- Correct valves numbers would be added (Comments 1 and 2),
- Figure 3-4 will be updated to add a note of interface regarding the orifice (will refer to proper figure in the Design Control Document [DCD]) (Comment 3),
- Figure 3-4 will be revised to change the dotted box to red and a description regarding the Eductor water storage tank would be added (Comment 4), and
- WCAP would be updated to reflect that there is no longer (as compared to the AP600) an interface between the demineralized water system and the primary sampling system (it was discussed that the paragraph that begins on the bottom of page 3-12 and Figure 3-5 would be deleted) (Comment 6).

It was agreed that no changes were necessary, at this time, in response to Comments 5 and 7, i.e., the NRC staff has sufficient docketed information to continue its review.

<u>WCAP-15985, "AP1000 Implementation of the Regulatory Treatment of Non[-]safety-Related Systems [RTNSS] Process"</u> (discussed on January 13 and 16, 2003)

In response to the comments, Westinghouse representatives stated that WCAP-15985 would be revised as follows:

- Update Table 1.1 in WCAP to include 7 additional systems (Comment 1),
- add startup feedwater system (SFW) to the list of 'front-line systems" (Comment 2),
- Address basis regarding contributions from external events and low power/shutdown events (Comment 3),
- Typographical errors identified in Comment 5 would be corrected,
- Westinghouse to verify compliance with GDC 2 (Comment 8),
- Section 10.2 would be revised to address Comment 9
- Revise WCAP to clarify issues documented in Attachment 2 (Comments 10, A and B).

It was agreed that no changes to the WCAP were necessary, at this time, in response to Comments 4 (although the NRC staff will investigate the response to RAI 720.057 due to possible relevant information) and 7. In addition, Westinghouse representatives stated that they will review Comment 6 and update the WCAP, if necessary.

<u>WCAP-15992, "AP1000 Adverse System Interactions Evaluation Report"</u> (discussed on January 16, 2003)

In response to the comments, Westinghouse representatives stated that WCAP-15992 would be revised as follows:

– Delete the last paragraph on page 2-36 (Comment 2),
– Revise WCAP to discuss regarding Westinghouse's position on credible single failure of squib valve (Comment 3),
– Revise WCAP and DCD to address Comment 4,
– Update WCAP and probabilistic risk assessment (PRA) to address Comment 5, and
– Revise WCAP to correct identified typographical errors (Comment 6).

No changes to the WCAP are necessary in response to Comment 1.

RAI 440.014 (discussed on January 13, 2003)

The NRC staff's comments were the same as were discussed in a call with Westinghouse on November 20, 2002 (see call summary dated December 12, 2002, ADAMS Accession No. ML023370461). Westinghouse representatives stated that a revised RAI response would be submitted to address the issues.

RAI 440.022 (discussed on February 13, 2003)

A. Westinghouse stated that they will revise its requested RAI response to provide the sensitivity factor values.

B. Westinghouse stated that they will revise its requested RAI response to provide the sensitivity factor values. In addition, Westinghouse stated that it would revise its response to provide the proprietary information not included in the DCD (WCAP-11397).

C. Westinghouse stated they would revise its RAI response to address the issues raised by the staff.

D. Westinghouse stated that they would revise its RAI response to identify items in Chapter 16 (capture under COL action item).

E. Westinghouse stated that they would revise its RAI response to address this issue.

RAI 440.037 (discussed on January 13, 2003)

Westinghouse representatives stated that they would verify adherence to the guidance of Regulatory Guide (RG) 1.190 (in DCD Section 5.3.2.6.2.2) and revise the DCD, if necessary.

RAI 440.043 (discussed on January 13, 2003)

Westinghouse representatives stated that they would submit a revised RAI response to address the issue.

RAI 440.048 and 440.049 (discussed on January 16, 2003)

Westinghouse representatives stated that the RAI responses and DCD would be revised to address the NRC staff's comments.

RAI 440.050 (discussed on January 13, 2003)

Westinghouse stated that Figure 6.3-3 in the DCD would be revised to address the NRC staff's comments.

RAI 440.052 (discussed on January 16, 2003)

Westinghouse stated that the RAI response and the DCD would be revised to address the NRC staff's comments.

Question on passive residual heat removal (PRHR) Heat Exchanger (HX) tube rupture:

Westinghouse will revise the DCD to address this issue.

RAI 440.076 (discussed on January 16, 2003)

Westinghouse stated that the RAI response would be revised to discuss preliminary runs.

RAI 440.077 (discussed on January 16, 2003)

Westinghouse to revise the RAI response to address the basis of the assumption that the loss of alternating current occurs at 2 seconds.

RAI 440.089 (discussed on February 13, 2003)

Westinghouse stated they would revise its RAI response to address this issue. In addition, the DCD would also be revised as necessary.

RAI 440.091 (discussed on January 13, 2003)

Westinghouse representatives stated that they would take the comments in Attachment 2 and inform us of how they intend to address the issues.

RAI 440.092 (discussed on January 13, 2003)

Westinghouse representatives stated that the RAI response would be revised to include design features or an appropriate reference regarding the flooding limit.

RAI 440.097 (discussed on January 13 and 29, 2003)

With respect to Comment A, Westinghouse representatives stated that a revised RAI response would be submitted which will include a summary of the calculation results that demonstrate compliance with the limitations discussed in Attachment 2.

With respect to Comments B, C, and D, Westinghouse representatives stated that a revised RAI response would be submitted to address the comments.

The NRC staff determined that no additional information was required for Comment E.

RAI 440.099 (discussed on January 13, 2003)

Westinghouse representatives stated that a revised RAI response would be submitted to address the issue of potential boron dilution.

RAI 440.102 (discussed on January 13, 2003)

The NRC staff determined that based on the information submitted to date, no further information from Westinghouse was necessary.

RAI 440.106 (discussed on February 13, 2003)

As stated in Paragraph 2 of Item B of the response to RAI-440-106, the minimum flow of 10,000 gallons per minute (gpm) (less than a full reactor coolant pump (RCP) flow) required in Limiting Condition for Operation (LCO) 3.4.9 assures adequate mixing of the reactor coolant system (RCS) in a boron dilution event. Item A of the RAI response indicates that the required flow rates are based on NUREG/CR--2733, “Experimental Data Report for LOFT Boron Dilution Experiment L6-6.” The LOFT test data shows that a residual heat removal (RHR) flow of 3,000 gpm provides a sufficient flow for adequate mixing of the fluid in the reactor vessel in the LOFT facility. The staff notes that the LOFT test was conducted in a small scaled test facility. It is not clear how the LOFT test data are applicable to the AP1000 in deriving the minimum required flow for supporting the perfect mixing model used in the analysis for the AP1000. The staff also notes that the minimum flow requirement for the AP1000 deviates from the basis for the required flow for the AP600, which requires (by LCO 3.4.9) at least one full RCP flow (about 52,000 gpm)

(A) Explain how the required minimum flow of 10,000 gpm is derived. If LOFT data are used, explain how the effect of the geometry differences in the LOFT facility and AP1000, and the scaling factors for both systems are determined.

(B) Justify why the required flow for the AP1000 deviating from the approved AP600 design is acceptable.

RAI 440.119 (discussed on January 13 and 16, 2003)

Westinghouse representatives stated that they would assess the need to clarify DCD Chapter 19E regarding the description of the inadvertent automatic depressurization system (ADS) actuation event.

440.183 and 440.184 (discussed on January 29, 2003)

At the time of the call these were new RAIs, i.e., they had not yet been issued in a letter to Westinghouse. The NRC staff discussed the issues raised in the RAIs. These RAIs were issue in a letter dated February 5, 2003.

RAI 720.005 (discussed on January 13, 2003)

(Note: this issue is associated with the liquid entrainment issue.) Westinghouse representatives stated that a revised RAI response would be submitted.

RAI 720.009 (discussed on January 16, 2003)

Westinghouse representatives stated that a revised RAI response would be submitted to address the issue.

RAI 720.010 (discussed on January 16, 2003)

NRC representatives stated that the comments that were discussed on November 20, 2002, were still valid and that the NRC staff was awaiting the revised RAI response (see telephone conference call summary dated December 12, 2002, ADAMS Accession No. ML023370461).

RAI 720.012 (discussed on January 13 and 16, 2003)

Westinghouse representatives stated that a revised RAI response would be submitted to address items A, B, C, and F. The NRC staff determined that no additional information was necessary for items D and E.

RAI 720.013 (discussed on January 16, 2003)

Westinghouse representatives stated that a revised RAI response would be submitted to address the issues. Westinghouse will revise the long-term cooling success criteria and correct typographical errors.

RAI 720.017 (discussed on January 16, 2003)

Resolution of this issue is tied to the resolution of RAI 720.009 and 720.013. Westinghouse to revise response to these RAIs.

RAI 720.021 (discussed on January 16, 2003)

Westinghouse representatives stated that a revised RAI response would be submitted to address the issues.

RAI 720.024 (discussed on January 16, 2003)

Westinghouse representatives will provide a revised RAI response to address the items listed for pages 6-12, 6-21, and 6-34. The issues identified for pages 6-25, 26, and 27 are tied to comments on RAI 720.021.

RAI 720.025 (discussed on January 13, 2003)

Westinghouse stated they would revise its RAI response to address these issues.

Docket No. 52-006

Attachment: As stated

RAI 720.025 (discussed on January 13, 2003)

Westinghouse stated they would revise its RAI response to address these issues.

Docket No. 52-006

Attachment: As stated

Distribution:

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JANUARY 13, 16, 29, AND FEBRUARY 13, 2003
TELEPHONE CONFERENCE CALLS SUMMARY
LIST OF PARTICIPANTS

| Nuclear Regulatory Commission | Westinghouse |
|---|---|
| Larry Burkhart | Mike Corletti |
| Gene Hsii | Tim Meneely |
| Tony Attard | Terry Schulz |
| Lambros Lois | Selim Sancaktar |
| Walt Jensen | Ed Cummins |
| Summer Sun | Kathy Demetri |
| Steve Bajorek | Ed Carlin |
| Joe Colaccinno | Bob Kemper |
| Joelle Starefos | George Roberts |
| | Andre Gagnon |
| | Rick Wright |
| | Jim Scobel |
| | Debra Ohkawa |
| | Erin Whiting |
| | Jean Luc Foret |
| | Bill Brown |

Attachment 1

NUCLEAR REGULATORY COMMISSION STAFF
COMMENTS THAT WERE SENT TO WESTINGHOUSE
TO FACILITATE DISCUSSIONS OF THE RAI RESPONSES
ON JANUARY 13, 16, 29, AND FEBRUARY 13, 2003

RAI 440.014

Regarding the response to question 440.014, the anticipated transient without scram (ATWS) event. The second to last sentence of the last paragraph, makes reference to Appendix A of the probabilistic risk assessment (PRA) and that the reactor coolant system (RCS) pressure is less than 3200 pounds-per-square inch (psig) with an unfavorable exposure time (UET) of 0. In reviewing Appendix A of the PRA report, it is not clear how the two statements agree. Please provide clarification.

RAI 440.022 - RTDP Design DNBR Limits:

(A) The response to Item A stated that the magnitude of the departure from nucleate boiling ratio (DNBR) sensitivities to the various parameters is dependent upon the conditions analyzed, and the calculations that are associated with the core limit conditions gave higher DNBR design limits than those associated with the other RTDP conditions.

Provide the sensitivity factor values of each parameter included in the RTDP process for the core limit conditions and the other RTDP conditions.

(B) In response to item B, Westinghouse provided the sigma values for the uncertainties of various parameters. Westinghouse did not provide the derivations of the design DNBR limits, nor the sensitivity factors of the parameters. Using the sensitivity factors in the sensitivity factor values of the sample calculation in Table 3-1 of WCAP-11397-P-A, and the sigma values Westinghouse provided, the RTDP design DNBR limits would be 1.226 and 1.215 for the typical cell and thimble cell, respectively, compared to 1.22 and 1.21, respectively, stated in the DCD.

Are the sensitivity factors in WCAP-11397 sample calculations used in the AP1000 RTDP design DNBR limits calculations? What are the bases for using these sensitivity factors, which were based on the THINC-IV code and the WRB-1 correlation, to the AP1000 design using the VIPRE-01 code and the WRB-2M correlation?

(C) The staff safety evaluation report (SER) approving WCAP-11397 included seven restrictions, e.g., inclusion of sensitivity factors in the safety analysis report, and re-evaluation of the sensitivity factors for any changes in departure from nucleate boiling (DNB) correlation, thermal-hydraulic code, or parameter values, etc.

Provide your evaluation that the use of the RTDP process for the AP1000 design complies with the seven restrictions in the WCAP-11397 SER.

(D) Westinghouse stated that the uncertainties values for power, $T_{in}$, pressure, and flow were assumed typical bounding values, and that the calculations will be revised when the plant is built.

Would this be a COL interface item? If the revised calculation results in higher design DNBR limits, will it invalidate the Chapter 15 safety analyses? What are the safety analysis DNBR limits used for the AP1000?

(E) The response to item C stated that the instrumentation uncertainty methodology will be similar to that used for the AP600 in WCAP-14605.

Provide reference to the methodology used for AP1000. Has it been approved?

RAI 440.037

The Westinghouse (W) response did not address the values of the co-variances in the adjustment process. However, in March 2001 the staff issued Regulatory Guide (RG) 1.190 which addresses fluence methodology.

Please verify that future vessel fluence calculations will adhere to the guidance of RG 1.190.

RAI 440.043 - Steam Generator Tube Rupture (SGTR)

Response to item (A) states that the AP1000 passive systems provide a unique response to the SGTR initiating event with respect to conventional plants by automatically terminating the loss of reactor coolant without depressurizing the RCS or overfilling the steam generator (SG).

Since the RCS is actually depressurized through the passive residual heat removal heat exchanger (PRHR HX) operation in response to an SGTR in the AP1000, clarify the statement ".... without depressurizing the RCS..."

RAI 440.048-049 - RVHVS [reactor vessel head vent system]

In the RCS ITAAC Table 2.1.2-4, Acceptance Criterion for Design Commitment, Item 8.e, (i.e., the RCS provides emergency letdown via the reactor vessel head vent system during design basis events to prevent long-term pressurizer overfill) is that the capacity of the reactor vessel head vent is sufficient to pass not less than 8.2 lbm/sec at 1250 psia in the RCS. This acceptance criterion is the same as the AP600.

Explain how this criterion is derived. Is it calculated from the RVHVS's venting capacity design basis of vent 40 percent of the RCS volume in one hour?

The response to 440.049 states that the design bases for the head vent include the ability to relieve water from the RCS at a flow sufficient to prevent pressurizer overfill during an event where the mass addition from the core makeup tanks (CMTs) causes an increase in the pressurizer inventory that otherwise might overfill the pressurizer.

What is the limiting transient for pressurizer overfill? What is the rate of insurge into the pressurizer? Is the ITAAC acceptance criterion of 8.2 lbm/sec corresponding to this insurge rate?

RAI 440.050 - In-containment refueling water storage tank (IRWST) injection line orifice

The response to RAI 440.050 explained the reason for not having an flow-adjusting orifice in the IRWST injection line. However, design control document (DCD) Figure 6.3-3 still shows an orifice in the IRWST injection line (Figure 6.3-2 does not show a flow orifice).

Will Figure 6.3-3 be revised?

RAI 440.052 - Automatic depressurization system (ADS) valves opening time

Table 15.6.5-7, "AP1000 ADS Parameters," is not clear regarding the values of various valve opening delay times (preset time delay and valve opening time), and the actual delay times used in Chapter 15 analysis.

What does Column 5 "valve opening time" mean? For example, stage 1 valve has opening time of ≤ 30 seconds. Does it mean it takes 30 seconds for the ADS-1 control (or depressurization) valve to be fully open? Is it the sum of the preset time delay after the opening of the isolation valve (see Section 7.3.1.2.4) plus the depressurization valve opening time? What are the values of the preset time delay and valve opening time? Also, does it mean that it takes 50 seconds (20 + 30) for the ADS-1 to be fully open?

Column 1, "Actuation Signal (Percentage of CMT level)," is not clear because of alignment. For example, for "Stage 1 - Control Low 1 67.5," does it mean that the first stage valves are opened 20 seconds after the CMT level reaches Low 1 (67.5 percent)? Also, Stage 4A does not have "control Low 2." Table 15.6.5-7 should be revised for clarity.

Question - PRHR HX tube rupture:

DCD Section 6.3.3.3.3 states that the PRHR HX tube rupture event is addressed in Section 15.6. Discuss where in Section 15.6 the event is addressed.

RAI 440.076

The last two sentences in the second paragraph of page 440.76-2 state that '[t]he resulting transient (a small break) would be less limiting than the double ended guillotine rupture of the larger feedline. This is confirmed by the results of sensitivity studies reported in WCAP-9230 and by preliminary runs performed to analyze the behavior of the AP1000 plant.'

Please summarize the results of the "preliminary runs" for the AP1000 design and address the adequacy of the quality of those "preliminary runs" used for supporting the AP1000 design certification application.

RAI 440.077

The last sentence of the fourth paragraph on page 440.077-1 states that "[t]his means that rod motion follows the time at which the double ended break open by two seconds,..."

Identify the contributors to the trip delay time of two seconds following the low SG level narrow range (NR) trip signal.

RAI 440.089 - Time of Power-Operated Relief Valve (PORV) Fails to Open Position

In the response to RAI 440-089, it stated that "... the PORV failure itself would result in an "S" signal being generated (on low steamline pressure, or low pressurizer pressure or level)..." This statement implies that the "S" signal can be actuated on a low pressurizer level signal. This is inconsistent with Item 1 of Technical Specification (TS) Table 3.3.2-1, which does not requires the pressurizer low level signal to actuate a "S" signal.

It also proposed to change DCD 15.6.3.2.1.2 and Table 15.6.3-1. Specifically, two changes are made to DCD 15.6.3.2.1.2 text as follows:

- change from "...to delay the low pressurizer pressure "S" signal..." to "to delay the low pressurizer pressure "S" and the low-2 pressurizer level signal..."

- change from " ... when the low-2 pressurizer level "S" signal is generated." to "...when either the low-2 pressurizer level signal or the low pressurizer "S" signal is generated."

These changes imply that the low-2 pressurizer level signal is not a "S" signal initiator.

However, Table 15.6.3-1 is revised from "Low pressurizer pressure "S" signal generated" to "Low-2 pressurizer level signal generated." According to the revised table, at 17 seconds (at 2515 seconds) after the initiation of the low-2 pressurizer level signal (at 2498 seconds), the CMT and PRHR operations begin. Following the requirements of items 2 and 13.e of TS Table 3.3.2-1, both CMT and PRHR will be actuated by a "S" signal. The change to Table 15.6.3-1 implies that the low-2 pressurizer level signal will actuate a "S" signal. The change is inconsistent with the changes in the DCD 15.6.3.2.1.2 discussed above.

Please clarify the identified inconsistences and make the necessary changes.

RAI 440.091

The response did not address the question i.e., given the differences in the physical dimensions between the AP600 and the AP1000 is WCOBRA/TRAC qualified to calculate liquid entrainment in the automatic depressurization system stage 4 (ADS-4) under conservative long-term cooling (LTC) conditions. The response does not allow a quantification of the liquid flow.

Please address the following: (1) the qualification of WCOBRA/TRAC to calculate liquid entrainment in the ADS-4 under conservative LTC conditions for the physical configuration of the AP1000, (i.e., slug flow in the AP600 vs droplet entrainment in the AP1000, (2) is 95 ft/sec above the entrainment velocity for the AP1000 configuration? and (3) quantify the liquid flow through ADS-4 for the conditions discussed in the December 2, 2002 response.

RAI 440.092

The calculation for wall-to-wall flooding is based on the assumption of a 9.0 gpm in-leakage rate. You conclude that "...core cooling would most likely not be demonstrated for the hypothetical case where wall-to-wall flooding was assumed immediately after the IRWST [in-containment refueling water storage tank] injection phase..."

Please address the basis for the assumption of a 9.0 gpm in-leakage under flooding conditions and that the walls of the dry spaces will not deform under the hydrostatic pressure.

RAI 440.097

440.097 (large-break loss-of-coolant accident [LBLOCA]):

In your response you state that "Reference 3 indicated the application restrictions on the AP600 methodology. The AP1000 large break LOCA analysis has complied with those restrictions." Because the estimated peak cladding temperature (PCT) is higher than 1725°F you must address limitation 4. You indicate that you addressed the global model matrix (Limitation 4a) and the sensitivity to the modeling of the CMT and PRHR (limitation 4b).

(A) Please address limitations 4c and 4d, i.e., maximum local oxidation and submit the results for staff review.

(B) The integrals of the flows shown in Figures 15.6.5A-5 and -6 do not match the contents of the accumulators and the CMTs, respectively, shown in Table 2.1-1 of WCAP-15612. Please comment on the flows shown in these figures and indicate in a single graph the core coolant inflow, outflow, downcomer level and core level vs time from the initiation of the transient to the initiation of in-containment refueling water storage tank (IRWST) injection (a similar graph was prepared for the AP600).

(C) What is the assumed single failure in the LBLOCA analysis and what is the exact calculated value of the maximum cladding oxidation? (Table 15.6.5-8)

(D) It is stated (page 15.6-46a) that "Figure 15.6.5A-12 presents the collapsed liquid levels in the core referenced to the bottom elevation of the active fuel (solid line) and downcomer (dashed line) referenced to the bottom of the reactor vessel." How is it possible that these levels have the same value at about 20 seconds?

(E) Best estimate LBLOCA analyses assume an unfavorable flow location for the hot assembly (even if the actual hot assembly is not in such a location in the configuration being analyzed). Was this implemented in this analysis?

RAI 440.099

Regarding the response to question 440.099 (boron dilution due to a small break loss-of-coolant accident [SBLOCA]), the sentence in the middle of the sixth paragraph states that the "lower AP1000 core boron concentration significantly reduces the potential to dilute the coolant in the reactor vessel to the point of criticality." Please provide additional clarification to this statement and the overall paragraph.

RAI 440.102

Regarding the response to question 440.102, associated with the use of the BEACON system, it is to be made clear that only those detectors that were reviewed as part of the original and supplement topical reports are to be used as part of the BEACON system.

RAI 440.106 - Required Minimum RCS Flow to Validate the Perfect Mixing Model Used in the Boron Dilution Event (DCD 15.4.5)

As stated in paragraph 2 of Item B of the response to RAI-440-106, the minimum flow of 10,000 gpm (less than a full RCP flow) required in LCO 3.4.9 assures adequate mixing of the RCS in a boron dilution event. Item A of the RAI response indicates that the required flow rates are based on NUREG/CR--2733, "Experimental Data Report for LOFT Boron Dilution Experiment L6-6." The LOFT test data shows that an RHR flow of 3,000 gpm provides a sufficient flow for adequate mixing of the fluid in the reactor vessel in the LOFT facility. The staff notes that the LOFT test was conducted in a small scaled test facility. It is not clear how the LOFT test data are applicable to the AP1000 in deriving the minimum required flow for supporting the perfect mixing model used in the analysis for the AP1000. The staff also notes that the minimum flow requirement for the AP1000 deviates from the basis for the required flow for the AP600, which requires (by LCO 3.4.9) at least one full RCP flow (about 52,000 gpm)

(A) Explain how the required minimum flow of 10,000 gpm is derived. If LOFT data are used, explain how the effect of the geometry differences in the LOFT facility and AP1000, and the scaling factors for both systems are determined.

(B) Justify why the required flow for the AP1000 deviating from the approved AP600 design is acceptable.

RAI 440.119 - Shutdown Evaluation

The response to RAI 440.119 indicates that the probability of a pipe break is significantly reduced at lower RCS pressure and temperature conditions in Mode 4 and below, and therefore, LOCAs are not postulated in these lower modes as design-basis accidents.

Specify the ranges of pressure and temperature for operation of Mode 4 and below and justify a LOCA (including RCS drainage resulting from a unisolated RCS pressure boundary such as pressure safety valve stuck open, unisolated paths to the letdown and charging system or residual heat removal system [RNS] system, or inadvertent opening of ADS) is not postulated to occur at lower Mode conditions.

440.183 (New RAI to be sent to Westinghouse via letter in separate correspondence)

In Section 4.3.2.7.3, Prediction of the Core Stability, the first paragraph makes reference to the fact that the AP1000's 14-foot core is slightly less stable with respect to axial xenon oscillations than the 12-foot cores. The same paragraph points to supporting information provided in subsequent Section 4.3.2.7.4 for explaining this increase in instability. But this section only provides information regarding 12-foot cores. No data is provided in support of a 14-foot core such as that from South Texas Plant, Units 1 and 2. Please provide additional technical justification in support of the 14-foot core for the increased axial xenon stability.

PRA Success Criteria (G. Hsii comments, see also W. Jensen comments below)

RAI 720.005

It is stated that the models in the MAAP4 code do not explicitly model in detail the upper plenum and hot leg entrainment.

Since the staff identified possible deficiencies in the NOTRUMP entrainment models at the time of ADS-4 actuation for the AP1000, why is the AP600 MAAP4/NOTRUMP benchmark applicable to the AP1000? Have you performed MAAP4/NOTRUMP benchmark for the AP1000? If not, how do you conclude that (response to item C) the limitations identified in WCAP-14869 for the AP600 apply to the AP1000 MAAP4 model and no additional limitations have been identified?

RAI 720.009

1. P1 last line: It states that "Figure 720.009-1 shows that MAAP4 PRHR HX removes a similar, but small amount of heat from the RCS as a function of pressure."

   Should "as a function of pressure" be "as a function of time?"

2. P. 2: The 3rd and 4th paragraphs, respectively, reference Attachment A and response to RAI 440.107.

   Should RAI 440.107 be RAI 440.054?

RAI 720.010, Page 2, 2nd paragraph

With regard to the spurious opening of four ADS-4 valves, the response states that the addition of a peak cladding temperature (PCT) uncertainty of 228 $^{o}$F to the WCOBRA/TRAC analysis result should bound the thermal-hydraulic uncertainty associated with this scenario. Since 228 $^{o}$F is the uncertainty for the double-ended cold-leg guillotine (DECLG), the response provides justification for its application to the spurious opening of four ADS-4 valves. The staff does not disagree with Westinghouse's justification based on the fact that the calculated PCT for this event is only 833 $^{o}$F, which has so much margin to 2200 $^{o}$F.

RAI 720.012, Attachment 1 - Summary of AP1000 T/H (Thermal-Hydraulic) Uncertainty Evaluation

A. Section 1.2, Item 6 states that the risk important sequences are identified using the acceptance criteria defined in Section 1.3. Section 3.1 states that " ... the acceptance criteria in Section 1.3 are used."

   Clarify where in Section 1.3 the acceptance criteria are defined. Would the "acceptance criteria" mean to be the screening criteria for dominant sequences defined in Section 1.4?

B. Table 4-1 lists the UC sequences analyzed for T/H uncertainty evaluation to bound the risk important sequences listed in Table 3.4. For long-term cooling cases, Case F indicates containment isolation and case G indicates no CI. (Section 4.1, Case F also states that containment isolation is available, and Case G indicates the containment pressure at atmospheric pressure.) (Response to RAI 720.013 also states that for Case F the containment is isolated, and Case G considers that the containment isolation failed).

   Clarify if these containment isolation situations are reversed from the risk important cases.

C. The long-term cooling Cases F and G assume one CMT is available.

   Justify these cases bound the Cases 3, 4, 5, 10, 12 and 13 in Table 3-4, where both CMTs have failed.

D. For short term cases, the emergency core cooling system (ECCS) availability selection may be bounding for the cases it intends to bound, but the initiating event may not be bounding. For example, Case C is said to bound risk important sequences 1, 7, 9,11.

   What is the justification for the DE DVI line break to bound other medium LOCA cases?

E. Section 4.1, Case E, Spurious ADS 4 large LOCA:

   It states that the case analyzed has successful containment isolation, and that analysis of the same case with failure of containment isolation is considered unnecessary because of very low PCT. However, Table 4-1 shows "no" containment isolation for Case E. Explain.

F. There are several typos:

   Table 3-2 "End State" column; UC should be "UC4."
   Section 4.1, Case D, 4th line, and Case E, line 6: "Event" should be "even."

RAI 720.013, Attachment

A. For long-term cooling Cases F and G, the availability of 1/1 CMT (A) is assumed.

   Are these cases bounding the 6 cases, listed in Table 3-3 and 3-4 in Attachment 1 to RAI 720-012, where both CMTs fail?

B. Section 1 states that it is conservatively assumed that the DEDVI line break occurs in the PXS-B room, since the size of this room is bigger than PXS-A. In Section 4.1, 3rd paragraph, it is stated that "initially, the only injection comes from the IRWST [in-containment refueling water storage tank] into the reactor vessel through the intact DVI [direct vessel injection injection] line (Fig. 720.013-14). Since ..... the level in the PXS-B room is below the DVI injection nozzle elevation, only steam from downcomer is vented out through the break (Fig. RAI 720.023-13 [TYPO]." However, Figure 720.013-14 shows DVI-B mixture flow rate, and Figure 720-013-13 shows DVI-A mixture flow rate. There appears to be discrepancies on the break location.

   Clarify whether the break is in DVI-A or DVI-B.

C. Same question for Section 4.2, Figures 720.013-14 and -13.

D. Section 3, "Methodology Implementation," states that WCOBRA/TRAC predicts higher ADS-4 flows, resulting in better depressurization of the primary system; consequently, the predicted IRWST injection rates were higher when using WCOBRA/TRAC than MAAP4. It was estimated that the IRWST would reach its lowest level about 2 hours earlier than as predicted by MAAP4. The IRWST level calculated by MAAP4 was adjusted to account for the more rapid draining predicted by WCOBRA/TRAC.

   Explain how the adjustment is made for MAAP4, and whether this adjustment is generic to all long-term cooling calculation cases, and provide justifications.

Following are Walt Jensen's comments on the PRA success criteria RAIs

RAI 720.009

Section A3.3.1 discusses success paths involving manual ADS-4 leading to in-containment refueling water storage tank (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." Case Nos. 1 and 2 assume manual actuation of ADS-4; however Case Nos. 1 and 2 assume credit for 4 of 4 ADS-4 valves and Case 2 takes credit for elevated containment pressure. Furthermore, Case 2 assumes credit for 2 of 2 accumulators. Please provide T/H uncertainty evaluation for the success paths in Section A3.3.1.

Open Issue: In RAI 720.009A the staff requested verification for the success paths identified in Section A3. These success paths list minimum sets of equipment necessary to maintain core cooling following postulated reactor system breaks. The minimum sets of equipment were identified using the MAAP4 computer code which has not been submitted by Westinghouse for NRC staff review. We therefore requested that Westinghouse include the minimum equipment success paths identified using the MAAP4 code in the T/H uncertainty analyses in Section A5. The analyses in Section A5 were performed using the NOTRUMP code which the NRC staff is now reviewing for application to the AP1000 safety analysis. We require that long-term core cooling be verified for the minimum sets of equipment listed in Section A.3.3.1. Long-term cooling was analyzed using WCOBRA/TRAC for 4 of 4 ADS-4 valves but not for 3 of 4 ADS-4 valves with containment isolation failure. See the response to RAI 720.31 dated December 2, 2002. (This was discussed with Westinghouse in a telephone conference call on November 20, 2002, see call summary dated December 12, 2002, ADAMS Accession No. ML023370461.)

RAI 720.010

Section A3.4 (also Section A5.1, LBLOCA 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 peak cladding temperature (PCT) for this equivalent of an LBLOCA event caused by the spurious opening of all ADS-4 valves with only one accumulator available is less severe than a double-ended (DE) 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.

Open Issue: The response to RAI 720.010 is inconsistent. Three cases are presented for LBLOCA uncertainty analysis.

Case No. 1 is the design-basis double-ended (DE) LOCA DCD. These results are given in Table 15.6.5-8 (RAI 440.97) to be a best estimate PCT of 184°F and a 95 percent value for PCT of 2124 °F. Case No. 2 is stated to be a split break. Table 15.6.5-7 gives a PCT of 1538 °F for breaks of this type. What do you mean by the following statement: "The ADS-4 valves do not have the same uncertainty with respect to break type, orientation?"

Case No. 3 is stated to be the same as Case No. 1 except that containment isolation is not assumed. No PCT is calculated because of the very large margin to 2200 °F. A PCT of 1060 °F is stated to have been calculated for Case No. 1. This is in conflict with Table 15.6.5-8. (See the response to RAI 440.097). The staff requires that appropriate analyses be provided to prove that the conditions postulated for Case No. 3 are indeed a success path. (This was discussed with Westinghouse in the telephone conference call on November 20, 2002. Westinghouse to clarify.)

RAI 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 AP600 T/H uncertainty analysis cases listed in Table A5.1-1. Many of these 15 AP600 cases are determined not to be 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 DBA-like analyses to demonstrate the acceptability of the PRA success criteria.

Open Issue: Table 720.012-1 indicates that a T/H uncertainty case has been run for the sequence of a DE DVI line break that includes among other failures failure of the PRHR (Case c). Such a sequence has not been included in the uncertainty cases of Appendix A to the PRA. A similar sequence was run using NOTRUMP (Case UC3). This case assumed that the PRHR heat exchanger functioned. See Table A5.2-1. (Reissued with RAI 720.015 response dated November 1, 2002.)

Open Issue: Table 3-1 lists UC (uncertain result) sequences with the predicted core damage frequency (CDF) and local release frequency (LRF). Thirteen of the sequences are bounded by NOTRUMP and WCOBRA/TRAC analyses. Those sequences that are calculated to have a low CDF or low LRF are not bounded. Those sequences that are not bounded both in the short-term and during long-term cooling have no approved analytical basis for success and should not be considered success paths in the PRA.

RAI 720.012 (Additional comments from Walt Jensen sent to Westinghouse via electronic mail on February 11, 2003)

The NRC staff stated previously to Westinghouse that they could not claim 3 ADS-4 without containment isolation to be a success. They said that was no problem since those sequences were all low risk. Please provide the following clarifications:

1. The response to RAI 720.012 states that success path OK6 is DB (design basis) ADS/CI fails. We understand that design basis ADS is only 3 ADS-4. Justify that those sequences that have an end state of OK6 are sufficiently low risk so as not to be significant for the PRA.

2. Success path OK7 is stated to be two accumulators/DB LLOCA (low loss-of-coolant accident). Westinghouse uses this success path to justify sequences with CMT failure. CMT failure is not design basis. These sequences should not be considered success for LLOCA.

3. Success path OK8 is stated to be an SI line break with an Auto ADS from faulted CMT. Westinghouse uses this success path to justify LLOCA sequences with containment isolation failure and three ADS-4 and CMT failures. These sequences should not be considered success for LLOCA.

4. Success path OK4 is stated to be More ADS-4 / Less ADS-1,2,3 than DB. Westinghouse uses this success path to justify core makeup tank large break (CMTLB) with 3 ADS-4 and no containment isolation. These sequences should not be considered success for CMTLB.

RAI 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.

Open issue: WCOBRA/TRAC boundary conditions including the interaction between the AP1000 primary system, passive safety systems, containment, and containment systems are stated to be calculated with MAAP. The NRC staff has not reviewed and approved the MAAP code for this purpose. (See open issues involving RAI 720.021 below). Furthermore in the response to RAI 440.009 (dated September 12, 2001), Westinghouse indicated that MAAP would not be used with WCOBRA/TRAC for long-term cooling analysis. This issue will remain open until Westinghouse provides long-term uncertainty analyses using approved methodology.

RAI 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.

Open Issue: The equipment analyzed for small break LOCA Case No. 3 cannot be considered a success unless long-term cooling is evaluated. Long-term cooling has been evaluated for three of four ADS-4 valves without containment failure and for four of four ADS-4 valves with containment failure. (See response to RAI 720.013 dated December 2, 2002.) Case No. 3 postulates three of four ADS-4 valves with containment failure. Since long-term cooling analyses have not been preformed with methodology that the staff has reviewed, Case No. 3

cannot be considered a success path. Section A.3.2.1 which lists the minimum equipment being credited for successful core cooling for the AP1000 also indicates that long-term cooling can be accomplished with three of four ADS-4 valves and failure of the containment to isolate. Long-term cooling uncertainty analyses should be provided using these assumptions.

RAI 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.

Open Issue: The response referenced an analysis using MAAP4 for which an 18-inch heating, ventilation, and air conditioning (HVAC) penetration was assumed to have failed opened. The NRC staff has not reviewed and approved the MAAP4 code for calculation of vapor and entrained droplet flow through a failed containment penetration. We require that the analysis be performed using a code that has been reviewed by the NRC staff. Assumptions for liquid and vapor flow from the failed containment penetration should be presented for staff review. The analysis was performed for a double ended DVI line which contains a 4-inch flow restrictor. Would sufficient water be retained within the containment for core cooling following larger break sizes?

RAI 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.

Open Issue: Success has not been demonstrated for the following failures:

P 6-3, 6-4: PRHR fails CMT success **but** three of four ADS-4 **and either** one of four ADS 2,3 valves **or** manual protection monitoring system (PMS) or diverse actuation system (DAS), Section 3.3 and P 6-21 of Table 1 indicate that PRHR is needed.

P 6-12: LLOCA or SPADS with one of two CMTs Section A3.4 is referenced which refers to Chapter 15 which assumes two CMTs.

P 6.21: Add MLOCA to PRHR Success cases. See page A-18

P 6-25, 26, 27: Water recirculating to the RCS with failure of containment isolation. See comments regarding response to RAI 720.021 dated November 26, 2002.

P 6-34: Where is the sensitivity study that one of two startup feedwater pumps delivering flow to two SGs is success following an anticipated transient without scram (ATWS).

There are 12 references to the AP600 PRA sections as the basis for determining success. For each reference discuss why the AP600 sections should be concluded to show success for the AP1000. Are these actual analyses or discussions of the systems? Give the AP600 analytical results and justify that they are applicable to the AP1000 or else re-analyze.

RAI 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 an SBLOCA, 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 the 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.

Open Issue: The table attached to the response states that at least 30 minutes of core cooling is available following an SLOCA, SG tuber rupture (SGTR), or Transient with no CMT or accumulator injection. This conclusion is based on calculations done using MAAP for the AP600 and MAAP calculations with one CMT for the AP1000. Plant-specific calculations should be provided for the AP1000 since the AP1000 has a higher power density than the AP600. Uncertainty analyses using NOTRUMP should be provided.

Other Issues:

The response to RAI 720.026 (October 18, 2002) refers to evaluations of chemical and volume control system (CVS) operation on PRHR operation that was done as part of the AP600 testing program. A reference should be provided. (Westinghouse agreed to provide such a reference in a telephone call of November 20, 2002)

The following are questions related to various topical reports that were submitted by Westinghouse on December 2, 2002.

<u>WCAP-15985, "AP1000 Implementation of the Regulatory Treatment of Nonsafety-Related System [RTNSS] Process"</u>

1. Table 1.1 (P.1.6):

The table lists non-safety systems evaluated in AP1000 RTNSS process in alphabetical order. It ends at steam generator.

Are there other non-safety systems after steam generator evaluated with the RTNSS process?

2. Table 2-1 (p. 2-4):

Table 2-1 lists only five non-safety systems and functions failed in PRA sensitivity studies, i.e., CVS, RNS, ECS, DAS and hydrogen ignitors.

Does that mean other non-safety systems (such as the plant control system, component cooling water system, etc.) not on this list are assumed to be operational in the PRA sensitivity studies? If so, are they RTNSS important? Would they be subject to TS or short-term availability administrative control?

3. Section 2 states that the PRA sensitivity studies are based on the AP1000 baseline PRA, and that they include an evaluation of internal events that occur at-power.

   Are external events and low power/ shut-down internal events included in the sensitivity studies?

4. Section 2.2 states that the PRA CDF and LRF - with assumed failure of the non-safety-related mitigation functions of the nonsafety-related SSCs - are reported in Chapter 50 of the AP1000 PRA report. However, Chapter 50 provides the importance and sensitivity analysis on CDF only.

   Clarify where in Chapter 50 describes the sensitivity studies for LRF?

5. Table 2-2 lists AP600/AP1000 PRA results for “baseline” and “without non-nuclear safety SSCs.” The AP1000 shutdown, internal event, baseline PRA CDFs are listed as 1.2E-7 and 1.2E-8 in Columns 2 and 3 for the shutdown internal events with and without manual DAS controls, respectively.

   Explain why the CDF for shutdown internal events without manual DAS controls is 10 times higher with manual DAS controls, whereas the LRF are the same for both cases.

6. Several places in Section 3, quote Section 59 of the PRA as the source of various initiating event contributions to the LRF, e.g., Sections 3.2 and 3.4 state both the RCS leak initiating and main steamline break event contributes only 1.5 percent to the LRF, Section 3.5 lists LRF for various events, etc.

   Clarify where in Chapter 59 these LRFs for various events are provided.

7. Section 2.3 described the DAS manual controls credited in the PRA mitigation functions which result in the LRF being within the safety goal. The DAS manual controls include the containment recirculation isolation valves.

   Since the containment recirculation isolation valves are a normally open MOV and a squib valve in series, and a check valve and a squib valve in series, describe how the DAS manual control for the containment recirculation isolation valves works.

8. Section 6 describes Post 72-hour Actions.

   Is the design of onsite equipment needed for post 72-hour support actions consistent with the GDC 2 requirement for protection against natural phenomena? Where is this documented?

9. Section 10.2.2 describes the missions of various non-safety-related plant systems, including the RNS for providing shutdown decay heat removal during RCS open shutdown conditions.

   Where is the RNS injection function to provide injecting cask loading pit water into the RCS following ADS actuation described?

10. Table 10-2, Investment Protection Short-term Availability Controls:

    A. 2.5 PCCWST and Spent Fuel Pool Makeup - Long-term Shutdown (P. 10-35):

       The ”Operability” for Item 2.5 states that Long-term makeup to the PCCWST should be operable.

       Since the PCS recirculation pumps provide the capability to transfer water from the PCS ancillary water storage tank to the PCS water storage tank and spent fuel pool for post 72-hour support actions, why does the “Operability” require only the makeup to the PCCWST, but not SFP, to be operable?

Clarify the basis for the PCS ancillary tank water volume of 725,000 gallons in SR 2.5.1 (whether it is for 7 days or a 3 day supply).

B. BASES for 3.3 AC power supplies - long-term shutdown (P. 10-56):

The initial water volume in the spent fuel pit normally provides for 3 days of spent fuel cooling, which differs from 7 days for the AP600. Explain the difference.

Clarify the basis for the ancillary fuel tank fuel volume of 600 gallons in SR 3.3.1 (3 or 7 days supply?).

<u>WCAP-15992, "Adverse System Interactions Evaluation Report"</u>

1. Section 2.2.1, Reactor Coolant Pump Interaction (P. 2-10, 2nd paragraph):

It states that "the AP1000 has addressed this (RCP - CMT) Interaction by providing a safety-related automatic trip of the RCPs on CMT actuation. In addition, <u>as stated earlier</u>, restart of the RCPs is not performed unless the CMT termination criteria can be met."

Please clarify where this was stated. Clarify what are the CMT termination criteria? What are the indications for the plant to have been brought to stable safe condition for CMT termination?

2. Section 2.3.1.5, CMT/Accumulator - ADS Interaction (P. 2-37):

It states that "in sensitivity studies performed to verify PRA success criteria, a phenomena was evaluated for cases of multiple failures in the ADS, where additional injection resulted in a higher (although acceptable) PCT. Ultimately, these multiple ADS failures were not credited as success in the PRA. ... additional CMTs or accumulators do not adversely impact the ability of the ADS to depressurize the RCS...."

Does that mean that the CMT and accumulator injection have no adverse interaction effect on the three out of four ADS-4 success criteria, but have potential adverse effect on the cases with less than three ADS-4 success? What are the effects on ADS 1,2, and 3 success criteria?

3. Section 2.3.4.1 Containment recirculation - PRHR HX interaction (P. 2-46):

Section 2.3.4.1 indicates that there is no significant interaction between containment recirculation and PRHR.

Since the containment recirculation paths contain a normally open MOV and a squib valve, could there be an adverse effect on the PRHR HX of a spurious opening of a squib valve, which could result in the IRWST draining to the containment?

4. P. 2-50, Section 2.3.5.5 PRHR HX - RCS Interaction

Would there be an adverse effect on the RCS piping and components of the PRHR HX operation that results in thermal stratification in the RCS?

5. P. 3-16, Item 2.3.6.4 ADS Actuation during Power Operation:

It states that a spurious actuation of the ADS during power operation would lead to a LOCA event. This event is explicitly modeled in the AP1000 PRA.

Do the PRA analyses cover both the spurious actuation of ADS-4 valves, spurious actuation of stages 1, 2 and 3 valves, or both?

6. Typos:

P.2-1, 4th paragraph, P. 2-29, 5th para., and P. 3-15, 3rd para.:
"AP600" should be "AP1000"?

P. 3-3, 1st para.: "In Section 4 ...." should be "In section 3.4...."?

P. 3-3, 3rd para.: "...in subsection 3.3.1..." should be "...in subsection 3.1 ..."?

Table 2-2, item 15.4.1 is repeated.

WCAP-15993, "Evaluation of AP1000 Conformance to Inter-System Loss-of-Coolant Accident [ISLOCA] Acceptance Criteria"

1. P. 3-4, Section 3.1.2, RNS Relief Valve, Line 7:

   The motor-operated CIV is listed as V04.
   Should it be V011 according to Fig. 3-1?

2. P. 3-9, Section 3.3.2, Line #6:

   CVS Makeup pump discharge line check valves are listed as V156A and B. (V156A and B are not check valves, but are isolation valves, according to Fig. 3-2).

   Should they should be V160A and B?

3. P. 3-11, Section 3.4.1 Primary Sampling System (PSS) Description, 2nd para. Lines 2 thru 6:

   It states that each connection of the PSS to RCS contains a flow-restricting orifice. However, the orifices are not shown in Fig. 3-4. Why?

4. Fig. 3-4 shows the only low-pressure components in PSS are Eductor water storage tank (EWST) and demineral water supply line.

   Are the Eductor supply pump seal, EWST drainage line, and EWST level indication line also low-pressure components?

   Explain the PSS design differences between AP600 and AP1000.

5. In Section 3.4.2, it states that “Even in the unlikely event that overpressurization would occur, leakage flow from the RCS would be well within the makeup capability of the normally operating makeup system.”

   Since the ISLOCA concern is LOCA outside containment, rather than makeup capability, the statement appears to be irrelevant.

6. Section 3.6.1 describes the demineralized water transfer and storage system interface with the primary sampling system. It references Figure 3-4, and discusses the demineralized water system (DWS) isolation valve V007, check valve V013, and isolation valve V037 (to the liquid waste system degasification). They are not shown in Figure 3.4. (Figure 3-4 and DCD Fig 9.3.3-1 do not show demineral water transfer system)

   Discuss and modify the figure, if necessary.

7. Section 3.6.2 states that a relief valve has been added to the DWS header inside containment to preclude the possibility of overpressurizing the DWS.

   Is the relief valve described in DCD Section 9.2.4?

<u>WCAP-15985, AP1000 Implementation of RTNSS</u>

1. Table 1.1 (P.1.6):

The table lists non-safety systems evaluated in the AP1000 RTNSS process in alphabetical order. It ends at steam generator.

Are there other non-safety systems after steam generator evaluated with the RTNSS process?

2. Table 2-1 (p. 2-4):

Table 2-1 lists only 5 non-safety systems and functions failed in PRA sensitivity studies, i.e., CVS, RNS, ECS, DAS and hydrogen ignitors.

Does that mean other non-safety systems (such as the plant control system, component cooling water system, etc.) not on this list are assumed to be operational in the PRA sensitivity studies? If so, are they RTNSS important? Would they be subject to TS or short-term availability administrative control?

3. Section 2 states that the PRA sensitivity studies are based on the AP1000 baseline PRA, and that they include an evaluation of internal events that occur at-power.

   Are external events and low power/shut-down internal events included in the sensitivity studies?

4. Section 2.2 states that the PRA CDF and LRF - with assumed failure of the non-safety-related mitigation functions of the nonsafety-related SSCs - are reported in Chapter 50 of the AP1000 PRA report. However, Chapter 50 provides the importance and sensitivity analysis on CDF only.

Clarify where in Chapter 50 describes the sensitivity studies for LRF?

5. Table 2-2 lists the AP600/AP1000 PRA results for “baseline” and “without non-nuclear safety SSCs.” The AP1000 shutdown, internal event, baseline PRA CDFs are listed as 1.2E-7 and 1.2E-8 in Columns 2 and 3 for the shutdown internal events with and without manual DAS controls, respectively.

   Explain why the CDF for shutdown internal events without manual DAS controls is 10 times higher than with manual DAS controls, whereas the LRF are the same for both cases.

6. Several places in Section 3, quote Section 59 of the PRA as the source of various initiating event contributions to the LRF, e.g., Sections 3.2 and 3.4 state that both the RCS leak initiating and main steamline break event contribute only 1.5 percent to the LRF, Section 3.5 lists LRF for various events, etc.

   Clarify where in Chapter 59 these LRFs for various events are provided.

7. Section 2.3 described the DAS manual controls credited in the PRA mitigation functions which result in the LRF being within the safety goal. The DAS manual controls include the containment recirculation isolation valves.

   Since the containment recirculation isolation valves are a normally open MOV and a squib valve in series, and a check valve and a squib valve in series, describe how the DAS manual control for the containment recirculation isolation valves works.

8. Section 6 describes Post 72-hour Actions.

   Is the design of onsite equipment needed for post 72-hour support actions consistent with the GDC 2 requirement for protection against natural phenomena? Where is this documented?

9. Section 10.2.2 describes the missions of various non-safety-related plant systems, including the RNS for providing shutdown decay heat removal during RCS open shutdown conditions.

   Where is the RNS injection function to provide injecting cask loading pit water into the RCS following ADS actuation described?

10. Table 10-2, Investment Protection Short-term Availability Controls:

    A. 2.5 PCCWST and Spent Fuel Pool Makeup - Long-term Shutdown (P. 10-35):

       The “Operability” for Item 2.5 states that Long-term makeup to the PCCWST should be operable.

       Since the PCS recirculation pumps provide the capability to transfer water from the PCS ancillary water storage tank to the PCS water storage tank and SFP for post

72-hour support actions, why does the “Operability” require only the makeup to the PCCWST, but not SFP, to be operable?

Clarify the basis for the PCS ancillary tank water volume of 725,000 gallons in SR 2.5.1 (whether it is for a 7 day or 3 day supply).

B. BASES for 3.3 AC power supplies - long-term shutdown (P. 10-56):

The initial water volume in the spent fuel pit normally provides for 3 days of spent fuel pool cooling, which differs from 7 days for the AP600. Explain the difference.

Clarify the basis for the ancillary fuel tank fuel volume of 600 gallons in SR 3.3.1 (3 day or 7 day supply?).

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Section 2.2.1, Reactor Coolant Pump Interaction (P. 2-10, 2nd paragraph):

It states that “the AP1000 has addressed this (RCP - CMT) Interaction by providing a safety-related automatic trip of the RCPs on CMT actuation. In addition, as stated earlier, restart of the RCPs is not performed unless the CMT termination criteria can be met.”

Please clarify where this was stated. Clarify what are the CMT termination criteria? What are the indications for the plant to have been brought to stable safe condition for CMT termination?

Section 2.3.1.5, CMT/Accumulator - ADS Interaction (P. 2-37):

It states that “in sensitivity studies performed to verify PRA success criteria, a phenomena was evaluated for cases of multiple failures in the ADS, where additional injection resulted in a higher (although acceptable) PCT. Ultimately, these multiple ADS failures were not credited as success in the PRA. ... additional CMTs or accumulators do not adversely impact the ability of the ADS to depressurize the RCS....”

Does that mean that the CMT and accumulator injection have no adverse interaction effect on the three out of four ADS-4 success criteria, but have potential adverse effect on the cases with less than 3 ADS-4 success? What are the effects on ADS 1,2, and 3 success criteria?

Section 2.3.4.1 Containment recirculation - PRHR HX interaction (P. 2-46):

Section 2.3.4.1 indicates that there is no significant interaction between containment recirculation and PRHR.

Since the containment recirculation paths contain a normally open MOV and a squib valve, could there be an adverse effect on the PRHR HX of a spurious opening of a squib valve, which could result in the IRWST draining to the containment?

P. 2-50, Section 2.3.5.5 PRHR HX - RCS Interaction

Would there be an adverse effect on the RCS piping and components of the PRHR HX operation that results in thermal stratification in the RCS?

P. 3-16, Item 2.3.6.4 ADS Actuation during Power Operation:

It states that a spurious actuation of the ADS during power operation would lead to a LOCA event. This event is explicitly modeled in the AP1000 PRA.

Do the PRA analyses cover both the spurious actuation of ADS-4 valves, spurious actuation of stages 1, 2 and 3 valves, or both?

Typos:

P.2-1, 4th paragraph, P. 2-29, 5th para., and P. 3-15, 3rd para.:
"AP600" should be "AP1000"?

P. 3-3, 1st para.: "In Section 4 ...." should be "In section 3.4...."?

P. 3-3, 3rd para.: "...in subsection 3.3.1..." should be "...in subsection 3.1 ..."?

Table 2-2, item 15.4.1 is repeated.

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